Institution of Washington will make the presentation.

C. *Meeting with the Director, NRC's Division of Waste Management, Office of Nuclear Materials Safety and Safeguards*—The Director will discuss items of current interest related to the Division of Waste Management programs which may include: Progress at the Yucca Mountain site, Status of Key Technical Issue resolution, and a discussion of shallow-land disposal long-term performance.

D. *Status of Nuclear Waste Related Research*—The Committee will meet with representatives of NRC's Offices of Nuclear Regulatory Research and Nuclear Material Safety and Safeguards to discuss the current status of nuclear waste related research.

E. *Preparation of ACNW Reports*—The Committee will discuss proposed reports, including: time frames for regulatory concern, the use of expert elicitation, elements of an adequate Low-Level Waste program, Committee priorities and task action plans, and biological effects from low-levels of ionizing radiation.

F. *Committee Activities/Future Agenda*—The Committee will consider topics proposed for future consideration by the full Committee and Working Groups. The Committee will discuss ACNW-related activities of individual members. The Committee will also consider potential new ACNW members. A portion of this session may be closed to public attendance to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(6).

G. *Miscellaneous*—The Committee will discuss miscellaneous matters related to the conduct of Committee activities and organizational activities and complete discussion of matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings were published in the Federal Register on September 27, 1995 (60 FR 49924). In accordance with these procedures, oral or written statements may be presented by members of the public 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 Committee, its consultants, and staff. Persons desiring to make oral statements should notify the Chief, Nuclear Waste Branch, Mr. Richard K. Major, as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting may be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief, Nuclear Waste Branch prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. Major if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Richard K. Major, Chief, Nuclear Waste Branch (telephone 301/415–7366), between 8 A.M. and 5 P.M. EDT.

ACNW meeting notices, meeting transcripts, and letter reports are now available on FedWorld from the ''NRC MAIN MENU.'' Direct Dial Access number to FedWorld is (800) 303–9672; the local direct dial number is 703–321–3339.

Dated: April 5, 1996.

John C. Hoyle,

Acting Advisory Committee Management Office.

[FR Doc. 96–8963 Filed 4–9–96; 8:45 am]

BILLING CODE 7590–01–M

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 16, 1996, through March 29, 1996. The last biweekly notice was published on March 27, 1996 (61 FR 13521).

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

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

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

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

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S.

Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

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

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

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

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

Date of amendment request: September 30, 1994, as supplemented September 18, 1995, January 19 and March 15, 1996

Description of amendment request: Currently, the steam generators (SGs) in place in the Catawba units are Westinghouse Model ''D'' type preheat SGs. The tube degradation levels in the SGs at Catawba Unit 1 have affected the reliability of the unit. Therefore, these generators are scheduled to be replaced with feedring SGs designed by Babcock

& Wilcox International. The design differences and analysis changes to support the feedring SGs result in the need to change the Technical Specifications (TS) in the following areas: (a) revise low-low SG water level for the reactor trip setpoint in TS Table 2.2-1 and for auxiliary feedwater actuation in TS Table 3.3-4, (b) revise high-high SG water level setpoint for turbine trip and feedwater isolation in TS Table 3.3-4, (c) delete reference to SG tube repair methods which will no longer be applicable after the replacement of the SGs and clarify initial surveillances, (d) revise reactor coolant system volume, (e) update Topical Report revision numbers in the Administrative Controls Section 6.9 of the TS, and (f) change the nominal average temperature in TS Table 2.2-1 for the reactor trip system setpoints to reflect the value incorporated into the safety analyses for the replacement SGs. The change made in the September 30, 1994, submittal, to reduce the steam line safety valve lift settings in TS Table 3.7-2, was withdrawn in the September 18, 1995, submittal. The January 19, 1996, submittal proposed changes to reflect the NRC's approved revisions to Topical Reports DPC-NE-3000 and DPC-NE-3002. The March 15, 1996, submittal provided additional information in response to NRC staff requests and also updated and clarified the involved TS pages including changes made to these TS pages by license amendments issued on other topics since the original application dated September 30, 1994.

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:

Operation of Catawba Unit 1 in accordance with the proposed changes to the Technical Specifications will not involve a significant increase in the probability or consequences of an accident previously evaluated. The low-low steam generator water level reactor trip setpoint, the high-high steam generator water level setpoint for turbine trip and feedwater isolation, and the low-low steam generator water level setpoint for auxiliary feedwater initiation are changing to support operation with the replacement steam generators. These setpoints were chosen both to optimize plant operation, and ensure that all applicable acceptance criteria are met for licensing basis safety analysis. These setpoints do not contribute to the initiation of any accident evaluated in the Catawba FSAR [Final Safety Analysis Report] and have no adverse impact on system operation, therefore it can be concluded that these changes will not significantly increase the probability or consequences of an accident evaluated in the FSAR.

The increase in Reactor Coolant System volume due to the replacement steam generators will not increase the probability or consequences of an accident previously evaluated. The increase in volume has no effect on the probability of occurrence of any accident evaluated in the FSAR. The mass and energy release inside containment due to postulated loss of coolant accidents inside containment has been analyzed to ensure that the peak containment pressure limit is not exceeded. All Chapter 15 reanalysis which was required due to the replacement steam generators assumed the new Reactor Coolant System volume. Since the results of these analyses show the applicable acceptance criteria continue to be met, it can be concluded that the consequences of an accident previously evaluated are not significantly increased due to this change.

Operation of Catawba Unit 1 in accordance with the proposed changes to the Technical Specification will not create the possibility of a new or different accident from any accident previously evaluated. The proposed changes to revise the low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater initiation ensure that the appropriate acceptance criteria for FSAR Chapter 15 transients which rely on these functions are met for operation with the replacement steam generators. ... The increase in Reactor Coolant System volume is taken into account in the analysis of the mass and energy release due to a postulated loss of coolant inside containment, and Chapter 15 events which have been reanalyzed due to replacement of the steam generators. As discussed above, the proposed changes will not introduce the possibility of a new or different accident from any previously evaluated, they will ensure that transients that take credit for these functions and dose analyses meet applicable acceptance criteria for operation with the replacement steam generators.

Operation of Catawba Unit 1 in accordance with the proposed changes to the Technical Specifications will not involve a significant reduction in a margin of safety. The proposed changes were made to ensure that transients that rely on low-low steam generator water level reactor trip setpoint, high-high steam generator water level setpoint for turbine trip and feedwater isolation, and low-low steam generator water level setpoint for auxiliary feedwater actuation meet applicable acceptance criteria. ... The proposed change in the Reactor Coolant System volume will not involve a significant reduction in a margin of safety. The increased volume affects the mass and energy release due to a postulated loss of coolant accident inside containment and the other Chapter 15 events which were reanalyzed due to replacement of the steam generators. This event has been analyzed and the results are within current acceptable limits. As discussed above, the acceptance criteria for FSAR transients which are affected by these proposed changes continue to be met, therefore there is no significant reduction in the margin of safety.

Changes to the steam generator surveillance requirements will simply delete inspection requirements which are no longer applicable after installation of the replacement steam generators. References to F* criteria, interim plugging criteria, and sleeving are deleted since these repair criteria were approved for use on the current steam generators. Since these changes only delete criteria which will no longer be applicable and cannot be used, no significant hazards considerations are involved.

The changes to Technical Specification 6.9.1.9 are administrative in nature. These changes are being made to reflect the most recent revisions of DPC-NE-3002 and DPC-NE-3000, which includes changes associated with the replacement steam generators. These topical report revisions [have been] reviewed and approved for use regarding McGuire and Catawba Nuclear Stations. Since these changes are administrative in nature, no significant hazards considerations are involved.

The proposed change to Technical Specifications [average coolant temperature in Table 2.2-1] does not involve a significant increase in the probability or consequences of an accident previously evaluated. Changing the value for [the average coolant temperature] in Notes 1 and 3 of Table 2.2-1 will update the value to agree with [the average coolant temperature] assumed in the applicable safety analyses for replacement of the steam generators. Acceptable results were obtained for all required reanalyses. The probability of an accident will not be significantly affected by operation with the new [average coolant temperature] value, because all equipment will be operated within acceptable design limits. The consequences of previously evaluated accidents which are affected by this change have been evaluated, and have been determined to be within acceptable limits.

This proposed change [to TS Table 2.2-1] will not create the possibility of a new or different kind of accident from any previously evaluated. This change does not change the physical configuration of the plant, and all analyses which are affected by replacement of the steam generators have been determined to have acceptable results assuming this value for [average coolant temperature].

This proposed change to the Technical Specifications [Table 2.2-1] will not involve a significant reduction in the margin of safety. All safety analyses which were affected by replacement of the steam generators assumed this value for [average coolant temperature] and the results were determined to be within previously acceptable limits.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: January 12, 1996, as supplemented March 4, 1996

Description of amendment request: This request was previously published in the Federal Register on January 31, 1996 (61 FR 3498). It is being renoticed to provide clarification to the scope of the original request. Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the allowable leakage rate values specified in the Technical Specifications (TS) and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On September 12, 1995, the NRC approved issuance of a revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B ‘‘Performance-Based Requirements’’ to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, ‘‘Performance-Based Containment Leak Test Program,’’ was developed as a method acceptable to the staff for implementing Option B. Accordingly, the licensee has submitted, in its application dated January 12, 1996, proposed changes to the TS to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide (RG) 1.163, ‘‘Performance-Based Containment Leakage-Test Program.’’ Although the licensee’s proposal indicated that it was consistent with RG 1.163, it did not include the clarifying changes to the TS that would require the visual examination of containment systems to be consistent with the guidance of RG 1.163. The licensee submitted a supplement, dated March 4, 1996, to its January 12, 1996 proposal, which proposes such changes to TS Surveillance Requirements 4.6.1.6 and 4.6.1.7 and 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 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 not previously evaluated.

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

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

The proposed changes are based on NRC-accepted provisions, and maintain necessary levels of reliability of containment integrity. The performanced-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 staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Herbert N. Berkow

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of amendment request: March 11, 1996

Description of amendment request: The proposed amendment would increase the alarm setpoints of the in-containment high range area and containment purge radiation monitors. These alarm setpoints are specified in Table 3.3-6 of Technical Specification 3.3.3.1. The proposed amendment would also include several editorial changes.

The proposed change to the in-containment high range area radiation monitor alarm setpoint would make the setpoint consistent with the Beaver Valley Power Station Emergency Action Levels (EALs) approved by the NRC in August 1994. These EALs use the in-containment high radiation area monitors as indication of fission product barrier challenges or failures.

The containment purge radiation monitors are provided to: (1) analyze the ventilation effluent from the reactor containment building, (2) detect abnormal releases and isolate the release if the setpoint is reached or exceeded, and (3) alert refueling personnel of the need to evacuate affected areas so as to maintain occupational exposures as low as reasonably achievable. The proposed increase in this setpoint value provides alarm and isolation based on offsite dose considerations and will provide greater operational flexibility since inadvertent engineered safety feature actuations due to evacuation alarms caused by minor (greater than three times background) increases in radiation levels will be minimized.

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

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

The proposed monitor alarm setpoint changes and editorial changes are administrative in nature. Should the in-containment high range area monitors fail to annunciate or give a false alarm, there would be no effect on any other plant equipment or systems. These monitors are safety related; however, they do not initiate any safety function, nor do they interface with any other safety related system. The monitors’ alarm as a visual (lighted icon) and audible alarm in the control room. The operator is then responsible for taking any corrective actions necessary, based on the alarm and Emergency Action Level (EAL) guidelines. The in-containment high range area monitors do not provide for any automatic actions of other equipment or systems when an alarm condition occurs.

The containment purge monitors are also safety related with the ability for an operator to input a radiation level value for high alarm levels during Mode 6, which upon actuation, create both a visual (lighted icon) and audible alarm in the control room. At the high alarm level, each monitor automatically sends a signal to close the purge supply and exhaust isolation dampers in the containment building. A change in the value of the alarm setpoint has no effect on the performance of the containment purge and exhaust system. The high alarm and subsequent automatic termination of a radioactive release will now be based on offsite dose considerations. There is no credible failure of the monitors associated with a change of the alarm setpoint value.

The operating and design parameters of the subject radiation monitors will not change. The proposed change affects only theradiation level at which an alarm condition is created and does not affect any accident assumptions. The in-containment high range area monitors' alarm setpoint change will not affect the radiological consequences of an accident. However, since the containment purge monitors revised setpoint is based on offsite doses consequences and is a higher value than the current setpoint of three times the background radiation level, the postulated offsite radiological consequences of a fuel handling accident inside containment would be increased. An analysis of a fuel handling accident inside containment with the purge and exhaust system discharging through the Supplementary Leak Collection and Release System (SLCRS) filter trains was performed and a summary of this analysis is to be added to Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). The analysis which determined the containment purge monitors' setpoint postulated offsite doses that are less than a small fraction (less than twenty-five percent) of the 10 CFR Part 100 guidelines. The fuel handling accident inside containment calculation demonstrated control room operator doses that comply with General Design Criteria (GDC) 19. Therefore, the increased radiological consequences of the change in the alarm setpoint are acceptable. The analysis assumed no isolation, so isolation actuated by the monitor alarm will reduce doses further.

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

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

The proposed radiation monitor alarm revisions cannot initiate a new type of accident. The referenced radiation monitors' alarms cannot initiate a new type of accident, since even a failure of the monitor itself cannot serve as the initiating event of an accident. Operator action is not made solely on a radiation monitor alarm; other plant condition indicators are also evaluated.

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

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

The in-containment high range area monitors have no capability to mitigate the consequences of an accident and do not interface with any safety related system. These monitors are safety related channels which provide indication to the operator of the integrity of the fission product barriers incontainment. This indication, combined with other indications of plant conditions may direct an operator to take action to mitigate the consequences of an accident. The alarm setpoint itself does not perform any specific safety related function and the trip value is not referenced in the UFSAR, nor does any site design basis document take credit for this setpoint. Safety limits and limiting safety system settings are not affected by this proposed change. The site will continue to meet the requirements of 10 CFR Part 100 which limits offsite dose following a postulated fission product release.

The containment purge monitors' revised setpoint is based on offsite dose consequences and is a higher value than the current setpoint of three times the background radiation level. Thus the postulated offsite radiological consequences of a fuel handling accident inside containment are increased which reduces the current margin of safety. An analysis of a fuel handling accident inside containment with the purge and exhaust system discharging through the SLCRS filter trains was performed and a summary of this analysis will be added to Chapter 15 of the UFSAR. The analysis postulated offsite doses to be less than twenty-five percent of the 10 CFR Part 100 guidelines and control room operator doses that comply with GDC 19. The analysis shows that the increased radiological consequences of the change in the alarm setpoint are acceptable. Further, the analysis assumed that no isolation would occur; therefore, isolation actuated by the monitors' alarm will reduce the postulated doses.

Therefore, use of the proposed technical specification would 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 1500l.

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

NRC Project Director: John F. Stolz

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

Date of amendment request: March 5, 1996

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) as follows:

(1) TS Surveillance Requirement (SR) 4.4.3.3: Delete the requirement for testing the switching capability for pressurizer heater power supplies on an 18-month interval.

(2) TS SR 4.5.2.d: Change the containment sump inspection requirements from each containment entry to once daily if a containment entry has been made and upon the final entry prior to establishing CONTAINMENT INTEGRITY.

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 the proposed amendments conform to the uidance given in Enclosure 1 of the NRC Generic Letter 93-05. The overall functional capabilities of the pressurizer heater system and the Emergency Core Cooling System (ECCS) will not be modified by the proposed changes. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons:

(1) Deleting the requirement to test the switching capabilities of the pressurizer heater emergency power supplies will reduce an unnecessary testing requirement since the pressurizer heaters are already connected to the emergency bus.

(2) Increasing the interval of containment sump inspections to once daily if containment has been entered and upon final entry will reduce unnecessary personnel exposure from performance of containment sump inspections for each containment entry.

[The staff notes that although statement (2) is correct, it does not provide a reason why the amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. The staff finds that once daily inspection of the containment adequately ensures that the containment sump remains free of debris.]

(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 proposed changes to the TS can not create the possibility of a new or different kind of accident from any accident 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 surveillance changes and inspection requirements, 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.

(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 are unchanged by the proposed amendments. The proposed changes to the TS which establish new or clarify old surveillance and inspection requirements [are] consistent with the NRC Generic Letter 93-05 line-item improvement guidance [and] do not significantly reduce any of the margins of safety even though the number of surveillances is decreased. These requested amendments are justified by the following reasoning from NUREG-1366:

(1) The surveillance or inspection results in radiation exposure to plant personnel which is not justified by the safety significance of the surveillances as in the case of the containment sump inspection requirements.

(2) The surveillance places an unnecessary burden on plant personnel because the time required is not justified by the safety significance of the surveillance as in the emergency power switching requirements for the pressurizer heater system.

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: Eugene V. Imbro

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

Date of amendment requests: February 22, 1996 (AEP:NRC:1243)

Description of amendment requests: The proposed amendments would revise the technical specifications to reference NRC Regulatory Guide 1.9, Revision 3 rather than NRC Regulatory Guide 1.108, Revision 1 criteria for the determination of a valid diesel generator test.

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

Per 10 CFR 50.92, proposed changes do not involve a significant hazards consideration if the changes do not:

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

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

3. involve a significant reduction in a margin of safety

Criterion 1

This amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change to the T/S [technical specifications] does not affect the assumptions, parameters, or results of any UFSAR [updated final safety analysis report] accident analysis.

The proposed amendment does not modify any existing equipment, and the proposed acceptance criteria for diesel generator testing will conform to NRC guidance. Based on these considerations, it is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed changes do not involve physical changes to the plant or changes in plant operating configuration. The proposed changes update guidance for diesel generator testing. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The proposed changes update guidance for the testing of diesel generators. The guidance is endorsed by the NRC in Regulatory Guide 1.9, and compliance with this guidance will ensure the operability of the diesel generators. Thus, 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 requests involve no significant hazards consideration.

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

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

NRC Project Director: Mark Reinhart, Acting

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

Date of amendment request: December 7, 1995

Description of amendment request: The proposed change will remove the requirement that primary containment always be purged or vented through the standby gas treatment (SBGT) system and adds requirements that would limit the use of SBGT for purging and venting. The proposed amendment also makes editorial changes and revises the associated Bases section.

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

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

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

The proposed change will allow primary containment to be purged or vented without the use of the SBGT system. This change only modifies the alignment of the atmospheric control system for purging or venting containment. The change does not affect any primary system, nor does it affect the ability of the containment isolation valves to close. As such, the proposed change can not affect the probability of occurrence of an accident previously analyzed. This change increases the possibility that some initial post-accident containment atmosphere could be released directly to the atmosphere at the top of the 375 foot stack prior to the closure of the containment isolation valves. However, this condition is bounded by the original radiological release analysis. This is balanced by the increased likelihood that post-accident reactor building atmosphere (from the time that the containment isolation valves close) is processed by the SBGT system.

The proposed technical specification also establishes strict controls for the use of the SBGT system for purging and venting containment atmosphere. This includes disabling the automatic initiation of the train not in use and relying on a dedicated operator to initiate the remaining train, should a DBA [design basis accident] occur. Since SBGT system operation does not affect the initiation of any postulated accident, disabling the automatic initiation and relying upon operator action to start the remaining train can not affect the probability of an accident previously evaluated. The failure of the train to start within one minute following the DBA could increase the consequences of an analyzed accident. To ensure timely initiation, NNECO has implemented a procedure for purging or venting through the SBGT system which establishes a dedicated operator whose function at the onset of a DBA is to isolate the train in use (the train expected to be damaged by the pressure spike), verify the open AC [atmospheric control] valves go closed, and then start the second train. This procedure has been validated to ensure that these actions can be completed within one minute.

Although not expected, a delay in operator action to initiate the SBGT has been evaluated for impact upon the radiological consequences. The evaluation shows that the offsite doses remain well within the 10CFR100 limit even if the operator actions are not completed until three minutes after the DBA occurs.

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

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

The proposed change allows removal of the SBGT system from the release path for normal containment purge and venting. The change does not affect the frequency or requirement for venting. Nor does the proposed LCO [limiting condition for operation] affect the processes of venting or purging primary containment; the same penetrations and containment isolation valves will continue to be used. All purging and venting functions can still be performed when required by existing specifications and plant procedures. The proposed change does not diminish the capability of any isolation valve for performing its isolation function.

Therefore, the proposed change can not create a new or different kind of accident.

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

The affect of this change has been analyzed against the criteria of 10CFR100 and 10CFR20. The potential release which may occur as a result of a postulated DBA while purging or venting directly to the stack will not exceed the limits of 10CFR100. Likewise, the technical specifications and administrative controls established for purging or venting through the SBGT minimize the potential for an unfiltered release should a DBA occur during that evolution. Further, the amount of time that a SBGT train is aligned to primary containment is expected to be substantially reduced from that required by the existing Technical Specification. Decreasing the amount of time that SBGT is aligned to primary containment decreases the possibility that a DBA would occur while in such an alignment.

Finally, the potential increase in dose which could occur as a result of normal purge and vent activities will be controlled such that it remains below acceptable limits.

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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment requests: January 17, 1996

Description of amendment requests: The proposed amendment would revise selected technical specifications (TS) in accordance with the NRC's Final Policy Statement on TS Improvements for Nuclear Power Reactors and relocate the TS to the Diablo Canyon Power Plant Equipment Control Guidelines. The proposed change would also create TS 6.8.4.j, ''Explosive Gas and Storage Tank Radioactivity Monitoring Program.'' Some of the TS would be relocated and maintained in accordance with this program. Specifically, the following TS would be relocated: TS 3.1.2.1, ''Boration Systems Flow Path - Shutdown,'' TS 3.1.2.3, ''Charging Pumps - Shutdown,'' TS 3.1.2.4, ''Charging Pumps - Operating,'' TS 3.1.2.5, ''Borated Water Sources - Shutdown,'' TS 3.1.2.6, ''Borated Water Sources - Operating,'' TS 3.3.3.2, ''Movable Incore Detectors,'' TS 3.3.3.4, ''Meteorological Instrumentation,'' TS 3.3.3.10, ''Explosive Gas Effluent Monitoring Instrumentation,'' TS 3.9.3, ''Decay Time,'' TS 3.9.5, ''Communications,'' TS 3.9.6, ''Manipulator Crane,'' TS 3.9.7, ''Crane Travel - Fuel Handling Building,'' TS 3.9.10.2, ''Water Level - Reactor Vessel - Control Rods,'' TS 3.9.13, ''Spent Fuel Shipping Cask Movement,'' TS 3.10.1, ''Special Test Exceptions - Shutdown Margin,'' TS 3.10.4, ''Position Indication System - Shutdown,'' TS 3.11.1.4, ''Liquid Holdup Tanks,'' TS 3.11.2.5, ''Explosive Gas Mixture,'' and TS 3.11.2.6, ''Gas Storage Tanks.''

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

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

The proposed changes simplify the Technical Specifications (TS), meet regulatory requirements for relocated TS, and implement the recommendations of the Commission's Final Policy Statement on TS Improvements and revised 10 CFR 50.36. Future changes to these requirements will be controlled by 10 CFR 50.59. The proposed changes are administrative in nature and do not involve any modifications to any plant equipment or affect plant operation.

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

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

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety-related system performs its function. Also, no changes to the operation of the plant or equipment are involved.

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 involve relocating TS requirements to a licensee-controlled document. The requirements to be relocated were identified by applying the criteria endorsed in the Commission's Final Policy Statement, which is included in the new revision of 10 CFR 50.36, and are consistent with NUREG-1431, Rev. 1. Thus, the proposed changes do not alter the basic regulatory requirements and do not affect any safety analysis.

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

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

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120

NRC Project Director: William H. Bateman

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

Date of amendment request: February 12, 1996

Description of amendment request: The amendments would revise the Susquehanna Units 1 and 2 Technical Specifications establish and reference a Primary Containment Leakage Rate Testing Program in order to implement 10 CFR 50, Appendix J, Option B in accordance with the guidelines contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program'', 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, which is presented below:

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

The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed license amendments revise the Technical Specifications to reflect the adoption of a performance-based containment leakage-testing program. The Nuclear Regulatory Commission has approved the use of a performance-based option for containment leakage testing programs when it amended 10 CFR Part 50, Appendix J (60 FR 49495).

To adopt of (sic) the revised regulations, licensees are required to incorporate into their Technical Specifications, by general reference, the NRC regulatory guide or other plant specific implementing document. A new Administrative Controls Specification is being added to the Susquehanna SES Technical Specifications that requires the establishment and maintenance of a Primary Containment Leakage Rate Testing Program. As stated in the Technical Specification, this Primary Containment Leakage Rate Testing Program will conform with NRC Regulatory Guide 1.163, ''Performance-Based Containment Leak-Rate Testing Program'', dated September 1995. The Primary Containment Leakage Rate Testing Program establishes requirements intended to ensure on-going containment integrity, including the performance of a periodic general visual inspection of the containment to detect early indications of structural deterioration.

The effect of increasing containment leakage rate testing intervals has been evaluated by the Nuclear Energy Institute using the methodology described in NUREG-1493 and historical representative industry leakage rate testing data. The results of this evaluation, as published in NEI 94-01, Revision 0, are that the increased risk corresponding to the extended test interval is small (less than 0.1 percent of total risk) and compares well to the guidance of the NRC's safety goal. The primary containment leak rate data and component performance history at Susquehanna SES are consistent with the conclusions reached in NUREG-1493 and NEI 94-01. Therefore, adoption of performance-based verification of leakage rates for isolation valves, containment penetrations, and the overall containment boundary will provide an equivalent level of safety and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No safety-related equipment, safety function, or plant operations will be altered as a result of the proposed license amendment.

The safety objective for the primary containment is stated in 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants.'' The safety function of the primary containment will be met since the containment will continue to provide ''an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment...'' for postulated accidents. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As stated above, the Nuclear Regulatory Commission has approved the use of a performance-based option for containment leakage testing programs when it amended 10 CFR Part 50, Appendix J (60 FR 49495). The new Primary Containment Leakage Rate Testing Program will conform with NRC Regulatory Guide 1.163, Revision 0, dated September 1995, ''Performance-Based Containment Leak-Rate Testing Program'' by requiring that leakage testing intervals be established based on the criteria in Section 11.0 of NEI 94-01, Revision 0.

As discussed in Part 1 above, the effect of increasing containment leakage rate testing intervals has been evaluated by the Nuclear Energy Institute using the methodology described in NUREG-1493 and historical representative industry leakage rate testing data. The results of this evaluation, as published in NEI 94-01, Revision 0, are that the increased safety risk corresponding to the extended test intervals is small (less than 0.1 percent of total risk) and compares well to the guidance of the NRC's safety goal. In addition, as demonstrated by risk analyses contained in NUREG-1482, relaxation of the integrated leak rate test frequency does not significantly increase the probability or consequences of a previously evaluated accident. Integrated leakage rate tests have been demonstrated to be of limited value in detecting significant leakages from penetrations and isolation valves. The primary containment leak rate data and component performance history at Susquehanna SES are consistent with the conclusions reached in NUREG-1493 and NEI 94-01. Therefore, the proposed license amendments adopting a performance-based approach for verification of leakage rates for isolation valves, containment penetrations, and the containment overall will continue to meet the regulatory goal of providing an essentially leak-tight containment boundary, will provide an equivalent level of safety, and do not involve a significant reduction in a margin of safety.

The revised Technical Specifications will continue to maintain the allowable leak rate (La) as the Type A test performance criterion. In addition, a requirement to perform a periodic general visual inspection of the containment is part of the performance-based leakage testing program.

The revised Technical Specifications will continue to maintain the allowable leak rate (La) as the Type B and C tests' performance criterion. As supported by the findings of NUREG-1493, the percentage of leakages detected only by integrated leak rate tests is small (only a few percent) and Type B and C leakage tests are capable of detecting more than 97 percent of containment leakages and virtually all such leakages are identified by local leak rate tests (LLRTs) of containment isolation valves.

Thus, the proposed license amendments do not involve a significant reduction in a margin of safety and will continue to support the regulatory goal of ensuring an essentially leak-tight containment boundary.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: February 23, 1996

Description of amendment request: The proposed amendment would change the Technical Specification (TS) Surveillance Requirement 4.6.2.1d concerning drywell-to suppression chamber bypass testing. Currently, Susquehanna TSs require the performance of a bypass test at 40 plus or minus 10-month intervals. The proposed TS change would request that the bypass test interval be revised to correspond with the interval for Primary Containment Integrated Leak Rate

Testing (ILRT) under 10 CFR Part 50, Appendix J, Option B.

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

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

The proposed change to allow bypass testing at the [Integrated Leak Rate Testing] interval involves no physical or operational changes to the Susquehanna SES. Reviews of bypass leakage test results at Susquehanna and other similarly designed plants confirm that minimal suppression pool bypass leakage has occurred. Based on this data, the risk of suppression pool bypass leakage from non vacuum breaker sources is no greater than that of other primary containment passive structures which are tested at the ILRT frequency. Leak testing of the drywell-to-suppression chamber vacuum breakers will continue to be performed on a refueling and inspection outage frequency to ensure that their contribution to the leakage area is acceptable. In addition, inspection of the diaphragm slab within the testing interval provides additional assurance that any degradation to the structure will be detected and resolved. Therefore, the pressure suppression capability of the containment is not reduced from the existing design, and there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to allow bypass testing at the ILRT interval involves no physical or operational changes to the Susquehanna SES. The surveillance change does not impact the LOCA response of the units, or impact the design basis of the units in any way. Therefore, the possibility of a new or different kind of accident will not be created.

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

The drywell-to-suppression chamber bypass leak test data obtained during previous testing at Susquehanna SES and other similarly designed plants demonstrates conformance, by a large margin, to the Technical Specification and design leakage requirements. The test data and safety analysis provided here indicate that there is negligible risk that the bypass leakage will change adversely in future years. Furthermore, the proposed performance based test methodology is judged to be acceptable based on the small risk of bypass leakage through paths other than those containing the suppression pool vacuum breakers. Testing of the bypass leak pathway containing the vacuum breakers will be used to verify acceptable bypass leakage during those outages when the bypass leak test is not performed. In addition, periodic visual inspection of the diaphragm slab within the bypass test interval provides additional assurance that any degradation to the structure will be detected and resolved.

Testing of the bypass leakage pathways containing vacuum breakers, with stringent acceptance criteria, combined with the other negligible potential leakage areas, and periodic inspection of the diaphragm slab, provide an acceptable level of assurance that the bypass leakage will be minimized. The proposed performance based approach to bypass testing and inspection ensures that adverse conditions can be detected and corrected such that the existing level of confidence that the primary containment will function as required during a LOCA is maintained. Therefore, the proposed Technical Specification changes do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: February 29, 1996

Description of amendment request: The proposed amendment relocates Technical Specification 3/4.9.6, ''Refueling Platform,'' to the Technical Requirements Manual, which is controlled under the requirements of 10 CFR 50.59.

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

The proposed change relocates the provisions of the Refueling Platform that are contained in the Technical Specifications and places them in the Technical Requirements Manual. Review and approval of those portions of the Refueling Platform requirements contained in the Technical Requirements Manual and revisions thereto will be the responsibility of the Plant Operations Review Committee just as it was their responsibility to review changes to the refueling platform Limiting Condition for Operation and Surveillance Requirements when they were part of the Technical Specifications. Requiring review by the Plant Operations Review Committee reinforces the importance of the Technical Requirements manual and the requirements controlled by it and assures a multidisciplined review. Approved Technical Requirements or changes thereto are provided to the Susquehanna Review Committee for information. No design basis accidents are affected by the change, nor are safety systems adversely affected by the change. Therefore, these changes will not result in any change to current Technical Specification requirements, but will reduce the level of regulatory control associated with the identified requirements. The level regulatory control has no impact on the probability or the consequences of an accident previously evaluated, therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates the provisions of the Refueling Platform that are contained in the Technical Specifications and places them in the Technical Requirements Manual. This change will not involve any physical changes to the Refueling Platform and its associated instrumentation nor any changes in the manner in which this equipment is operated, maintained, tested or inspected. Future changes to these relocated requirements or surveillances will be evaluated in accordance with the requirements of 10CFR50.59. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is not reduced. The relocated requirements do not meet any of the four criteria in the NRC Policy Statement used for defining the scope of Technical Specifications. In addition, the relocated requirements and surveillances for the refuel platform and associated instrumentation remain the same as stated in the existing Technical Specifications. Future changes to these relocated requirements or surveillances will be evaluated in accordance with the requirements of 10CFR50.59. Review and approval of those portions of the Refueling Platform requirements contained in the Technical Requirements Manual and the revisions thereto will be the responsibility of the Plant Operations Review Committee just as it was their responsibility to review changes to the refueling platform Limiting Condition for Operation and Surveillance Requirements when they were part of the Technical Specifications. Approved Technical Requirements or changes thereto are provided to the Susquehanna Review Committee for information. Therefore, no significant reduction in a margin to safety is proposed.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: February 29, 1996

Description of amendment request: The proposed amendment removes the Rod Block Monitor (RBM) requirements from the Technical Specifications, thereby reducing the number of rod movements during power maneuvers.

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 proposed change removes the Rod Block Monitor requirements from Technical Specifications based on no credit being taken for the RBM in the reload licensing analysis. The RBM was originally designed to prevent fuel damage during the Rod Withdrawal Error [RWE] event by automatically stopping control rod motion before any fuel design limits are exceeded. However, due to control rod drift events in which the RBM can not (sic) stop control rod motion, the RWE is analyzed without taking credit for the RBM. The results of this analysis are operating limits that prevent fuel damage from a RWE in which control rod motion is not stopped by the RBM. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change of removing the RBM requirements from Technical Specifications does not change the currently approved approach for performing the reload licensing analysis for either Unit. To date all reload analyses have been performed considering the rod drift event as a moderate frequency event and no credit being taken for the RBM. Since no credit is taken, removal of these requirements from Technical Specifications does not impact the current approach for performing reload analysis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Continued compliance to the governing General Design Criteria [GDC] for the RWE analysis assumes an appropriate margin of safety.

GDC 10 is met when the specified acceptable fuel design limits (SAFDLs) are not exceeded for the RWE. The first SAFDL requires that a MCPR [Minimum Critical Power Ratio] Operating Limit be determined such that the reduction of MCPR margin due to an RWE does not violate the MCPR Safety Limit. The second SAFDL requires that the uniform cladding strain does not exceed 1% during an RWE. PP&L's [Pennsylvania Power and Light Company] licensing analysis of the RWE, without taking credit for the RBM, determines a MCPR Operating limit such that the reduction of MCPR margin due to an RWE does not violate the MCPR Safety Limit and validates that the maximum uniform cladding strain is less than 1%. Therefore, the applicable SAFDLs for the RWE are satisfied and the GDC requirements met.

GDC 20 is met when the reactivity control system is automatically actuated to prevent exceeding the SAFDLs. PP&L's licensing analysis of the RWE, without taking credit for the RBM, conservatively determines a MCPR Operating Limit and validates that the maximum uniform cladding strain is less than 1%. Therefore, actuation of the RBM is not necessary to prevent exceeding the applicable SAFDLs for the RWE.

GDC 25 is met when a single malfunction in the reactivity control system will not cause the SAFDLs to be exceeded. The current RWE licensing analysis assumes a control rod drift event without any credit for the RBM. With respect to the reactivity control system, the assumptions of a control rod drift event and no actuation of the RBM are more conservative than the assumptions in the original SSES Safety Evaluation. Therefore, the requirements from GDC 25 are still met. Therefore, no significant reduction in the safety margin exists.

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

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

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

NRC Project Director: John F. Stolz

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: February 9, 1996, as supplemented March 15, 1996. This notice supersedes the notice published on February 28, 1996 (61 FR 7557) in its entirety.

Description of amendment request: The proposed amendment would revise the Administrative Controls Section 5.6.6 of the Ginna Technical Specifications (TSs) to incorporate a reference to the methodology for determining pressure/temperature (P/T) and low-temperature overpressure protection (LTOP) limits. The proposed amendment would follow guidance given in Generic Letter 96-03 for relocating LTOP and the reactor coolant system (RCS) P/T limits to the RCS Pressure and Temperature Limits Report (PTLR). The proposed amendment will allow the licensee to perform future LTOP and RCS P/T evaluations, using NRC-approved methodology, without requiring changes to the TS.

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

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes only require that future RCS P/T and LTOP limits be developed using NRC approved methodology as specified within the Administrative Controls section and do not involve any technical changes. As such, these changes are administrative in nature and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the changes do not impact any safety analysis assumptions other than requiring future evaluations of RCS P/T and LTOP limits to be performed in accordance with NRC approved methodology. These changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Acting Project Director: Susan Frant Shankman

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: March 19, 1996

Description of amendment request: The licensee is proposing to change the Technical Specification (TS) 3/4.2.4, QUADRANT POWER TILT RATIO (QPTR), the Bases for QPTR, and TS 3/4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION, Table 3.3-1, ''Table Notation, Action Statement 2.c.'' The licensee is requesting the changes in order to use the guidance in the improved Westinghouse Standardized Technical Specifications, NUREG 1431, Rev. 1.

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

1. The probability or consequences of an accident previously evaluated in the FSAR is not significantly increased.

The QPTR limits ensure that $F^{N}_{delta-H}$ and $F_{Q}(z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution. In MODE 1, the $F^{N}_{delta-H}$ and $F_{Q}(z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. The QPTR satisfies Criterion 2 of the NRC Policy Statement.

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the departure from nucleate boiling ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_{Q}(z)$ and $F^{N}_{delta-H}$ is possibly challenged. With the QPTR exceeding its limit, a power level reduction of 3% from RATED THERMAL POWER for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power.

The Power Range Neutron Flux trip setpoint reduction is not required since incore flux measurements are not expected to change concurrent with the loss of a Power Range Channel. These setpoints, which were previously reduced in order to account for uncertainties, will now be monitored and corrected, if necessary, per TS 3.2.4.

Any change in the QPTR would be detected by requiring a check of the QPTR once per 12 hours. If the QPTR indicates an increase, THERMAL POWER has to be reduced accordingly. A 12 hour completion time is sufficient because any additional change in QPTR would be relatively slow.

The improvement of TS 3/4.2.4 to reflect the improved STS in no way impacts the accident analysis of the FSAR. Therefore, the probability or consequences of a previously evaluated accident has not been increased.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The proposed amendment request does not necessitate physical alteration of the plant nor changes in parameters governing normal plant operation. Therefore, the change does not create the possibility of a new or different kind of accident or malfunction.

3. The margin of safety has not been significantly reduced.

This proposed amendment request precludes core power distributions that may lead to violation of the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm; and

d. The control rods must be capable of shutting down the reactor with a minimum required shutdown margin with the highest worth control rod stuck fully withdrawn.

The improvement of TS 3/4.2.4 ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses.

The core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This reevaluation is required to ensure that the reactor core conditions are consistent with the assumptions in the safety analyses. Therefore, the margin of safety has not decreased.

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: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: Frederick J. Hebdon

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: November 2, 1995

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.8.1, ''AC Sources - Operating,'' of the improved TS, to (1) extend the offsite circuit allowed outage time (AOT) from ''72 hours AND 6 days from discovery of failure to meet LCO'' to ''72 hours AND 10 days from discovery of failure to meet LCO'' and (2) extend the emergency diesel generator (EDG) AOT from ''72 hours AND 6 days from discovery of failure to meet LCO'' to ''7 days AND 10 days from discovery of failure to meet LCO.'' Additionally, the licensee proposes to further extend the EDG AOT to ''10 days AND 10 days from discovery of failure to meet LCO'' on a once-per-refueling cycle frequency for maintenance purposes.

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 Emergency Diesel Generators (EDGs) are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. EDGs are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The EDGs provide backup power to components that mitigate the consequences of accidents. The proposed changes to the Allowed Outage Times (AOTs) do not affect any of the assumptions used in the deterministic safety analysis.

To fully evaluate the effect of the EDG AOT extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of accidents previously evaluated.

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

This proposed change does not alter the design, configuration, or method of operation of the plant. Therefore, this 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 changes do not affect the Limiting Conditions for Operation or their Bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes, and these evaluations determined that the changes are either risk neutral or risk beneficial.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: November 6, 1995

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.5.1, ''Safety Injection Tanks (SITs),'' of the improved TS to extend, in general, the allowed outage time (AOT) for a single inoperable SIT from 1 hour to 24 hours. Additionally, the licensee proposes to extend the SIT AOT from 1 hour to 72 hours if a single SIT becomes inoperable due to malfunctioning SIT water level and/or nitrogen cover pressure instrumentation.

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

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

The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System (ECCS). The SITs are not accident initiators in any accident previously evaluated.

Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The SITs are designed to mitigate the consequences of Loss of Coolant Accidents (LOCAs). The proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Therefore, the consequences of accidents previously evaluated do not change.

To fully evaluate the SIT Allowed Outage Time (AOT) extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated.

The proposed change pertaining to SIT inoperability based solely on instrumentation malfunction does not involve a significant increase in the consequences of an accident as evaluated and endorsed by the Nuclear Regulatory Commission (NRC) in NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements.''

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

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this 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 changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes are either risk neutral or risk beneficial.

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 three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: November 8, 1995

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.5.2, ''ECCS - Operating,'' in the improved TS to extend the allowed outage time from 72 hours to 7 days for a single low pressure safety injection (LPSI) train.

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 Low Pressure Safety Injection (LPSI) system is a part of the Emergency Core Cooling System (ECCS). Inoperable LPSI components are not considered to be accident initiators. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The LPSI system is primarily designed to mitigate the consequences of a large Loss of Coolant Accident (LOCA). This proposed change does not affect any of the assumptions used in the deterministic LOCA analysis. Therefore, the consequences of accidents previously evaluated do not change.

To fully evaluate the LPSI Allowed Outage Time (AOT) extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated.

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

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this 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 affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes.

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 three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: December 6, 1995

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 4.3, ''Fuel Storage,'' of the improved TS, to allow fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent to be stored in both the spent fuel racks and the new fuel racks.

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.

There is no increase in the probability of an accident because the physical characteristics of a fuel assembly are not changed when fuel enrichment is increased. No changes will be made to any safety related equipment or systems. Fuel assembly movement will continue to be controlled by approved fuel handling procedures.

Fuel cycle designs will continue to be analyzed with Nuclear Regulatory Commission (NRC)-approved codes and methods to ensure the design bases for San Onofre Units 2 and 3 are satisfied.

The double contingency principle of American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 8.1-1983 can be applied to any postulated accident in the Spent Fuel Pool (SFP) which could cause reactivity to increase. In conjunction with administrative controls for heavy loads and impact zones, a boron concentration of 1850 parts per million (PPM) (the current Technical Specification (TS) limit) is sufficient to maintain k-eff less than or equal to 0.95 for all normal and postulated accident conditions.

Regarding the new fuel storage racks, there is no postulated accident which could cause reactivity to increase above 0.95 for all moderator densities from 0.0 to 1.0 grams/cubic centimeter (gms/cc).

The radiological consequence analyses performed in the Updated Final Safety Analysis Report (UFSAR) include the development of source terms which bound discharge fuel burnups to 60,000 megawatt days per ton (MWD/T). Increasing the San Onofre Units 2 and 3 enrichment to 4.8 weight percent (w/o) does not result in discharge fuel assembly burnups greater than 60,000 MWD/T. Thus, the consequences of the fuel handling accident are unchanged from the current UFSAR bases.

Therefore, this proposed 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 accident previously evaluated.

The proposed changes do not involve any physical changes to the plant or any changes to the method in which the plant is operated. They do not affect the performance or qualification of safety related equipment. Fuel handling accidents were previously considered. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

For the SFP, the NRC acceptance criteria is k-eff less than or equal to 0.95 under all normal and accident conditions and including uncertainties. For the new fuel storage racks, k-eff must remain less than 0.95 if completely flooded with unborated water, and must remain below 0.98 in an optimum moderation event. Analyses have been performed which demonstrate that these acceptance criteria will continue to be met when the enrichment is increased to 4.8 w/o.

The current UFSAR design bases SFP decay heat loads bound the proposed enrichment increase due to the reduced fuel batch size.

Radiological effects of fuel handling accidents are unchanged by this enrichment increase.

The proposed design of the higher enriched fuel will result in a slight weight increase. However, the seismic event is bounded by the analyses performed for the rerack project.

Therefore, there will not be a significant reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: January 4, 1996

Description of amendment requests: The licensee proposes to delete License Conditions 2.C(26) and 2.C(27). These license conditions require the licensee to implement and maintain a plan for scheduling all capital modifications based on an NRC approved Integrated Implementation Schedule Program Plan.

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 deletes an administrative means of tracking and scheduling NRC required plant modifications and license commitments. It does not affect the plant configuration nor NRC mandated schedules for implementation of modifications. Because the deletion of the license condition does not affect the plant configuration, no accident analyses are affected; therefore, the proposed change does not increase the probability or consequences of any previously evaluated accident.

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

The proposed change will not alter the configuration of the plant or its operation; therefore, the proposed change does not create 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 is administrative and does not affect any accident analyses or involve any modification to the plant configuration; therefore, the proposed change does not involve a reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Tennessee Valley Authority, Docket Nos. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: February 28, 1996

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to extend the ice weighing and flow channel inspection surveillance frequencies from 9 to 18 months. Concurrently, the required total ice bed weight would be increased from 2,360,875 to 2,403,800 lbs. to account for the anticipated additional ice sublimation during the longer interval between weighing and inspection.

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 analysis is presented below.

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

The ice condenser system is provided to absorb thermal energy release following a LOCA or high energy line break (HELB) and to limit the peak pressure inside containment. The containment analysis for Watts Bar is based on a minimum of 1093 lbs of ice per ice basket evenly distributed throughout the ice condenser, and the subcompartment analysis is based on 85 percent of the available flow area (flow channels) being open uniformly throughout the ice condenser. For the predicted sublimation rate of up to 12 percent for 18 months, an average ice basket weight of 1093 lbs at the end of the 18 month period would still be available. An evaluation of the operating history of the other operating ice condenser plants shows that after 18 months 85 percent of the flow channels will still be available.

Thus the ice condenser will perform its design functions with the revised minimum ice weight and inspection interval. There will be no design change or other operational changes. Accordingly, the proposed changes to the technical specifications do not affect the probability or consequences of an accident.

2.

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

As stated above, the proposed changes do not involve modifications to the ice condenser or other plant systems. Hence there is no possibility of a new or different kind of accident since no new design is involved.

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

Plant safety margins are established through limiting conditions of operation, limiting safety system settings, and safety limits specified in the TS. None of these will be changed.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

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

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-390 Watts Bar Nuclear Plant,Unit 1, Rhea County, Tennessee

Date of amendment request: February 28, 1996

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) surveillance frequency for Westinghouse type AR relays, used as solid state protection system slave relays or auxiliary relays, from quarterly to a refueling outage frequency.

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

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

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same ESFAS instrumentation is being used and the same ESFAS system reliability is expected. The proposed change will not modify any system interface or function and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent the performance of any accident mitigation systems or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the safety analysis report. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

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

This change does not alter the performance of the ESFAS mitigation systems assumed in the plant safety analysis. Changing the interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios. Implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total ESFAS system response assumed in the safety analysis. The periodic slave relay functional verification is relaxed because of the demonstrated high reliability of the relay and its insensitivity to any short term wear or aging effects. Implementation of the proposed amendment does not result in a reduction in a margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: March 12, 1996

Brief description of amendments: The proposed amendments would revise Technical Specification (TS) 3/4.6.1.1, ''Containment Integrity,'' 3/4.6.1.2, ''Containment Leakage,'' 3/4.6.1.3, ''Containment Air Locks,'' and 3/4.6.1.6, ''Containment Structural Integrity,'' and add new TS 6.8.3g, ''Containment Leakage Rate Testing Program,'' to implement the new performance-based leakage rate testing program as permitted by 10 CFR Part 50, Appendix J.

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

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

The proposed changes to the TS and the addition of specification 6.8.3g to implement the new performance based Containment Leakage Rate Testing Program, have no effect on plant operation. The proposed changes only provide mechanisms within the TS for implementing a performance based methodology for determining the frequency of leak rate testing which has been approved by the Commission. The test type and test method used for testing would not be changed. The test acceptance criteria would not be changed and containment leakage will continue to be maintained within the required limits.

Directly referencing the Containment Leakage Rate Testing Program for containment [integrated leak rate test] ILRT and [local leak rate test] LLRT requirements does not involve any modification to plant equipment or affect the operation or design basis of the containment. Leakage rate testing is not a precursor to or an initiating event for any accident.

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

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

The proposed changes only allow for the implementation of Option B testing frequencies and do not involve any modifications to any plant equipment or affect the operation or design basis of the containment. The proposed changes do not affect the response of the containment during a design basis accident.

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

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

The proposed changes do not adversely affect a Safety Limit, Limiting Condition for Operation (LCO) or plant operations. These changes only implement the allowed Option B testing frequencies that have been determined by the Commission not to involve a safety concern. The testing method, acceptance criteria and bases are not changed and still provide assurance that the containment will provide its intended function.

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

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

Local Public Document Room location: 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

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

Date of amendment request: March 21, 1996

Description of amendment request: The proposed changes clarify the requirements for testing the charcoal adsorbent in the auxiliary ventilation and control room air filtration systems as outlined in Technical Specifications 4.12 and 4.20, 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 probability or consequences of an accident previously evaluated is not significantly increased.

The charcoal testing clarifications and explict reference to the testing currently conducted do not affect system operation or performance, nor do they affect the probability of any event initiators. The changes do not affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. Therefore, the proposed changes do not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR [Updated Final Safety Analysis Report].

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The clarification to the charcoal sample testing protocol does not affect the method of operation of the system. The proposed changes clarify and explicitly identify the testing methodology for the charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated is not created by this change.

3. The margin of safety has not been significantly reduced.

The charcoal adsorber sample laboratory testing accurately demonstrates the required performance of the adsorbers following a design basis LOCA [loss-of-coolant accident] or Fuel Handling Accident. Changing the Technical Specifications to clarify the actual test methodology and explicitly [referencing] the charcoal testing actually performed does not affect system performance or operation. Therefore, these changes do not result in a significant reduction in any margin of safety.

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

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

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

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: February 19, 1996

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) Section 4.2 and its associated basis by allowing the application of a voltage-based repair limit for the steam generator tube support plate intersections experiencing outside diameter stress corrosion cracking. The proposed repair criteria are based on guidance provided in Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes affected by Outside Diameter Stress Corrosion Cracking,'' dated August 14, 1995, and on associated industry guidance.

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

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

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

Testing of model boiler specimens for free span tubing (no TSP [tube support plate] restraint) at room temperature conditions show burst pressures in excess of 5,000 psig for indications of ODSCC [outside diameter stress corrosion cracking] with voltage measurements as high as 19 volts. Burst testing performed on five intersections pulled from the Kewaunee SGs [steam generators] with up to a 2 volt indication showed measured tube burst in the range of 9,537 to 9,756 psig. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show burst pressures in excess of 6,300 psi at room temperatures. Correcting for the effects of temperature on material properties and the minimum strength levels, tube burst capability significantly exceeds the safety factor requirements of RG [Regulatory Guide] 1.121.

Tube burst criteria are inherently satisfied during normal operating conditions due to the presence of the TSP. Test data indicates that tube burst cannot occur within the TSP, even for tubes with through wall EDM [electro-discharge machining] notches 0.75 inch long, when the notch is adjacent to the TSP. Since tube burst is precluded during normal operating conditions, the criterion that must be satisfied to demonstrate adequate tube integrity is a safety margin of 1.43 times MSLB [main steam line break] pressure differential. The BOC [beginning of cycle] structural limit for 7/8 inch diameter tubing is 8.82 volts. Applying an allowance of 20.5% for NDE [nondestructive examination] uncertainty and 50% for crack growth rate over an operating cycle results in a voltage repair limit of 5.4 volts. The proposed repair limit of 2 volts is very conservative when compared to the 5.4 volts taking into account the low average growth rates experienced at Kewaunee and the high tube burst pressures.

Relative to the expected leakage during accident condition loadings, a plant specific calculation was performed to determine the maximum primary-to-secondary leakage during a postulated MSLB event. The evaluation considered both pre-accident and accident initiated iodine spikes. The results of the evaluation show that the accident spike yielded the limiting leak rate. This case was based on a 30 rem thyroid dose at the site boundary and initial primary and secondary coolant activity levels of 1.0 uCi/gm and 0.1 uCi/gm dose equivalent iodine-131, respectively. A leak rate of 34.0 gpm was determined to be the upper limit for allowable primary to secondary leakage in the SG in the faulted loop. The SG in the intact loop was assumed to leak at a rate of 0.1 gpm (150 gpd).

Application of the voltage-based repair limit will be supplemented with a projected EOC [end of cycle] MSLB leakage calculation and conditional burst probability assessment. The methodology for performing these calculations will be in accordance with the GL [generic letter]. Should the projected MSLB leakage be exceeded indications will be repaired or removed from service until the projected leakage is less than or equal to 34.0 gpm.

Application of the voltage-based repair limit will not adversely affect SG tube integrity. Therefore, the proposed amendment will not increase the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed voltage-based repair limit will not reduce the overall safety or functional requirements of the SG tube bundles. The tube burst criteria will be satisfied during normal operating conditions by the presence of the TSPs. The RG 1.121 criteria that must be satisfied during accident loading conditions is 1.43 times MSLB differential pressure. Conservatively, the existing data base of burst testing shows that the tube burst margins can be satisfied with bobbin coil signal amplitudes of about 8.82 volts or less regardless of the depth of tube wall penetration.

The proposed repair criteria will be supplemented with a reduced operating leakage requirement of 150 gpd through either SG to preclude the potential for excessive leakage during operating conditions. The 150 gpd restriction will provide for timely leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. The operating leakage limit is based on leak-before break considerations, critical crack length and predicted leakage.

The SG tube integrity will continue to be maintained through inservice inspections and primary-to-secondary leakage monitoring. Therefore, the proposed change will not create the possibility of a new or different kind or accident.

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

Application of the voltage-based repair criteria has been demonstrated to maintain tube integrity commensurate with the RG 1.121 criteria. RG 1.121 describes a method acceptable to the staff for meeting GDCs [general design criteria] 2, 14, 15, 31 and 32. This is accomplished by determining the limiting degradation of SG tubing as established by inservice inspection, beyond which tubes should be removed from service. Upon implementation of the repair criteria, even under the worst case conditions, the occurrence of ODSCC at the TSPs is not expected to lead to a SG tube rupture event during normal or faulted conditions. The most limiting event would be a potential increase in leakage during a MSLB event. Excessive leakage during a MSLB is precluded by verifying that the expected EOC crack distribution of ODSCC indications at TSP locations would result in an acceptably low primary-to-secondary leakage. Therefore, the radiological consequences from tubes remaining in service is a small fraction of the 10 CFR 100 limits.

The combined effects of a LOCA [loss-of-coolant accident] plus SSE [safe shutdown earthquake] on the SGs were assessed as required by GDC 2. This issue was addressed for the Kewaunee SGs through the application of leak-before-break (LBB) principles to the primary loop piping. Based on the results of this analysis, it is concluded that the LBB is applicable to the Kewaunee primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the primary loops, the LOCA loads from the large branch lines were also assessed and found to be of insufficient magnitude to result in SG tube collapse. Based on these analysis results, no tubes are expected to collapse or deform to the degree that the secondary-to-primary in-leakage would be increased over currently expected levels. On this basis no tubes need to be excluded from the voltage-based repair criteria for reasons of deformation resulting from combined LOCA or SSE loadings.

Addressing the RG 1.83 considerations, implementation of the voltage-based repair criteria will include a 100% bobbin coil probe inspection of all tube-to-TSP intersections with known ODSCC down to the lowest cold leg TSP identified. This will be supplemented by a reduced operating leakage limit, enhanced eddy current data analysis guidelines, MRPC [motorized rotating pancake coil] inspection requirements and a projected EOC voltage distribution. It is concluded that the proposed change will not result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: January 16, 1996

Brief description of amendments: The amendments revise the Technical Specifications to reflect approval of the use of 10 CFR Part 50, Appendix J, Option B, for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, containment leakage rate test program for Type A tests only.

Date of issuance: March 13, 1996

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

Amendment Nos.: 212 and 189

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5810) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 13, 1996.No significant hazards consideration comments received: No

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

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Calvert County, Maryland

Date of application for amendments: November 30, 1995, as supplemented by letter dated March 15, 1996.

Brief description of amendments: The amendments allow the installation of tube sleeves as an alternative to plugging for repair of steam generator (SG) tubes using repair techniques developed by Westinghouse Electric Corporation. The November 30, 1995, letter also requested approval of repair techniques developed by ABB Combustion Engineering, Inc., for repairing SG tubes. The NRC staff is still reviewing that portion of the request and will notice the results of its review at a future date.

Date of issuance: March 22, 1996

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

Amendment Nos.: 213 and 190

Facility Operating License No. DPR-53 and DPR-69: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 176) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 22, 1996.No significant hazards consideration comments received: No

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

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: November 22, 1995

Brief description of amendment: The proposed change will delete the qualifying statement, ''... provided the remaining systems are in continuous operation,'' from TS Section 3.3.4.2.

Date of issuance: March 15, 1996

Effective date: March 15, 1996

Amendment No. 168

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

Date of initial notice in Federal Register: December 6, l995 (60 FR 62487) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 1996No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 18, 1994, as supplemented by letters dated October 9, 1995, February 13 and March 8, 1996

Brief description of amendments: The amendments revise the current combined Technical Specifications (TS) for Units 1 and 2 by separating them into individual volumes for Unit 1 and Unit 2. In addition to the changes required by the TS split, some administrative and editorial changes were made, such as the correction of typographical errors and the deletion of unnecessary blank pages.

Date of issuance: March 21, 1996

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

Amendment Nos.: Unit 1 - 166 - Unit 2 - 148

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

Date of initial notice in Federal Register: September 14, 1994 (59 FR 47166) The October 9, 1995, February 13 and March 8, 1996, letters provided additional information that did not change the scope of the July 18, 1994, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 1996 and Environmental Assessment dated February 7, 1996. No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: December 15, 1995, as supplemented March 5, 1996

Brief description of amendments: These amendments (1) revise Technical Specifications (TSs) 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 3/4.6.1.6, and associated Bases, (2) delete TS 6.9.2.g, and (3) add a new TS 6.17. These changes make the TSs consistent with Option B of Appendix J of 10 CFR Part 50 and the implementing guidance of Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' dated September 1995. Option B of Appendix J permits implementation of a performance-based leak rate test schedule in lieu of the prescriptive requirements contained in Option A of Appendix J. These amendments remove from the TSs the prescriptive requirements of Option A concerning test frequencies and test methodology. These amendments also include minor administrative and editorial changes to add consistency between the Bases and the TSs and provide additional clarification.

Date of issuance: March 19, 1996

Effective date: Both units, as of the date of issuance, to be implemented within 60 days.

Amendment Nos.: 197 and 80

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 179) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: August 16, 1995

Brief description of amendment: This amendment modifies Technical Specification 3.6.6.1, Shield Building Ventilation System (SBVS), to more effectively address the design functions performed by the SBVS for both the Shield Building and the Fuel Handling Building.

Date of issuance: March 20, 1996

Effective date: March 20, 1996

Amendment No.: 81

Facility Operating License No. NPF-16: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49937) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 10, 1995, as supplemented on December 21, 1995, and February 22, 1996

Brief description of amendment: The amendment deletes a Technical Specification (TS) reference to the reactor trip input to the reactor building isolation system, changes the surveillance frequency for the sodium hydroxide storage tank and station battery, and removes an inappropriate reference in the TS bases section to testing that is not required by the TSs themselves.

Date of issuance: March 21, 1996

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

Amendment No.: 200

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58401). The December 21, 1995 and February 22, 1996 letters did not change the staff's determination hazards consideration exist. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 1996No significant hazards consideration comments received: No

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

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: December 19, 1995 and supplemented February 16, 1996 (AEP:NRC:1215B&D)

Brief description of amendments: The amendments modify the technical specifications to replace the existing scheduling requirements for overall integrated and local containment leakage rate testing with a requirement to perform the testing in accordance with 10 CFR Part 50, Appendix J, Option B. Option B allows test scheduling to be adjusted based on past performance.

Date of issuance: March 19, 1996

Effective date: March 19, 1996, with full implementation within 45 days

Amendment Nos.: Unit 1 - 209, Unit 2 -193

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1632) The February 16, 1996 supplement made only a minor change to the proposed technical specifications that provided consistency between the wording for Units 1 and 2. The change did not affect the staff's proposed finding that the amendments involve no significant hazards consideration.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 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.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: February 17, 1996

Brief description of amendment: The amendment allows one main steam line's leakage rate to be as high as 35 standard cubic feet per hour (scfh) as long as the total leakage through all four main steam lines does not exceed 100 scfh until the end of Operating Cycle 6.

Date of issuance: March 18, 1996

Effective date: March 18, 1996

Amendment No.: 83

Facility Operating License No. NPF-58: This amendment revised the Technical Specifications. The Commission's related evaluation of the amendment and final no significant hazards consideration determination is contained in a Safety Evaluation dated March 18, 1996. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 7823 dated February 29, 1996). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 1, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

Date of amendment request: December 19, 1994 (TXX-94274), as supplemented by letter dated January 25, 1996 (TXX-96026)

Brief description of amendments: These changes allowed testing of Reactor Protection System and Engineered Safety Features Actuation System instrument channels with the channel under test in bypass in order to reduce the vulnerability to spurious trips during surveillance testing.

Date of issuance: March 14, 1996

Effective date: March 14, 1996

Amendment Nos.: Unit 1 - 47; Unit 2 - 33

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

Date of initial notice in Federal Register: February 1, 1995 (60 FR 6312) The additional information contained in the supplemental letter dated January 25, 1996, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 1996. No significant hazards consideration comments received: No.

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

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

Date of amendment requests: November 21, 1995 (TXX-95289), as supplemented by letters dated February 22 (TXX-96061 and TXX-96062) and 28, (TXX-96068), and March 13, 1996 (TXX-96090).

Brief description of amendments: The amendments allowed both doors of the containment personnel airlock to be open during fuel movement and core alterations, providing one airlock door is capable of being closed and the water level in the refueling pool is maintained.

Date of issuance: March 18, 1996

Effective date: March 18, 1996

Amendment Nos.: Unit 1 - 48; Unit 2 - 34

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

Date of initial notice in Federal Register: January 3, 1996 (61 FR 185) The additional information contained in the supplemental letters dated February 22 (2 letters) and 28, and March 13, 1996, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 18, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 26, 1994, as supplemented January 5, April 25 and October 12, 1995, and February 2 and March 1, 1996.

Brief description of amendments: These amendments revise the Technical Specifications by extending the operation of both units with the current heatup and cooldown limit curves to 23.6 effective full power years. The basis for TS Section 15.3.1.B, ''Pressure/Temperature Limits,'' is also revised to reflect the methodology for the curve compilation.

Date of issuance: March 20, 1996

Effective date: March 20, 1995

Amendment Nos.: 168 and 172

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

Date of initial notice in Federal Register: July 20, 1994 (59 FR 37093). The supplemental submittals provided additional 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 March 20, 1996.No significant hazards consideration comments received: No.

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

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 10, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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

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

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

Date of application for amendment: March 23, 1996

Brief description of amendment: The amendment modifies Technical Specification (TS) 4.8.2.1.c, ''DC Sources - Operating,'' to specify that the provisions of TS 4.0.1 and 4.0.4 are not applicable. This provision expires upon entry into Mode 4 coming out of the sixth refueling outage or upon any deep discharge cycle of the battery.

Date of issuance: March 23, 1996

Effective date: March 23, 1996

Amendment No.: Unit 2 - 94

Facility Operating License No. NPF-51: The amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 23, 1996.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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

NRC Project Director: William H. Bateman

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

Date of application for amendment: March 26, 1996

Brief description of amendment: The amendment revises Technical Specification (TS) 3/4.9.6 to allow the refueling machine overload cutoff limit to be increased to as much as 2000 pounds, from the current 1600 pound limit, in an effort to free the stuck fuel assembly from core location A-06. The additional 400 pound increase will be applied in 50 pound increments. This change will expire when the fuel assembly located at core location A-06 is successfully withdrawn.

Date of issuance: March 26, 1996

Effective date: March 26, 1996, to be implemented prior to entry into Mode 4 from the current refueling outage.

Amendment No.: Unit 2 - 95

Facility Operating License No. NPF-51: The amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 26, 1996.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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

NRC Project Director: William H. Bateman

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of application for amendment: March 29, 1996

Brief description of amendment: The amendment clarifies the testing requirements and updates the regulatory and industry guidance references for charcoal adsorber units addressed by TS 4.6.4.4, Hydrogen Purge System; TS 4.6.5.1, Emergency Ventilation System; and TS 4.7.6.1, Control Room Emergency Ventilation System.

Date of issuance: March 29, 1996

Effective date: March 29, 1996

Amendment No.: 209

Facility Operating License No. NPF-3: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 29, 1996.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

Dated at Rockville, Maryland, this 3rd day of April 1996.

For the Nuclear Regulatory Commission.

Steven A. Varga,

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

[FR Doc. 96–8786 Filed 4–9–96; 8:45 am]

BILLING CODE 7590–01–F

All Licensees of Reactors With Installed Thermo-Lag Fire Barrier Material; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has acted on Petitions for action under 10 CFR 2.206 received by a letter dated September 26, 1994, from the Citizens for Fair Utility Regulation and the Nuclear Information and Resource Service; by a press release dated October 6, 1994, from the Maryland Safe Energy Coalition; by separate letters dated October 21, 1994, from the GE Stockholders' Alliance and Dr. D. K. Cinquemani; by a letter dated October 25, 1994, from the Toledo Coalition for Safe Energy; by a letter dated October 26, 1994, from R. Benjan; by a letter dated November 14, 1994, from B. DeBolt; and by a letter dated December 8, 1994, from the Nuclear Information and Resource Service and the Oyster Creek Nuclear Watch. The Petitioners requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to the use of Thermo-Lag by reactor licensees and that their letters be treated as Petitions pursuant to Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 CFR 2.206).

The Citizens for Fair Utility Regulation and the Nuclear Information and Resource Service requested (1) Texas Utilities Electric Company, the licensee of Comanche Peak Steam Electric Station, Unit 1, perform additional destructive analysis for Thermo-Lag configurations in proportion to the total installed amount to determine the degree of ''dry joint'' occurrence, (2) the licensee perform fire tests on upgraded ''dry joint'' Thermo-Lag configurations for conduit and cable trays to rate the barrier as a tested configuration in compliance with fire protection regulations, and (3) the NRC immediately suspend the Comanche Peak Unit 1 license until the above listed corrective actions are taken. The Maryland Safe Energy Coalition requested immediate shutdown of both reactors at the Peach Bottom plant until the risk of fire near electrical control cables due to combustible insulation is