The Subcommittee will review the Westinghouse methodology for best-estimate small-break loss of coolant accident analysis, using the WCOBRA/TRAC code. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

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

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

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

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

Dated: October 29, 1998.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 98–29500 Filed 11–3–98; 8:45 am]

BILLING CODE 7590–01–P

Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards; Joint Meeting of the ACRS Subcommittees on Reliability and Probabilistic Risk Assessment, Plant Operations, and on Regulatory Policies and Practices; Notice of Meeting

The ACRS Subcommittees on Reliability and Probabilistic Risk Assessment, Plant Operations, and on Regulatory Policies and Practices will hold a joint meeting on November 19 and 20, 1998, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, November 19, 1998—1:00 p.m. until the conclusion of business.

The Subcommittees will continue their discussion of proposed options to make 10 CFR Part 50 and 10 CFR 50.59 (Changes, tests and experiments) risk informed.

Friday, November 20, 1998—8:30 a.m. until the conclusion of business.

The Subcommittees will review proposed changes to the NRC assessment programs, including integrated review of assessment processes, Senior Management Meeting process, as well as risk-informed baseline inspection case studies and associated proposed changes to the inspection program, and other related risk-informed initiatives. The Subcommittees will also discuss AEOD programs for risk-based analysis of reactor operating experience including risk-informed performance indicators. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

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

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

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

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Michael T. Markley (telephone 301/415–6885) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: October 29, 1998.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 98–29501 Filed 11–3–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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 fromOctober 9, 1998, through October 23, 1998. The last biweekly notice was published on October 21, 1998 (63 FR 56238).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination, and Opportunity for a Hearing

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

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

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

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

By December 4, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's

Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of application for amendment: October 6, 1998.

Description of amendment request: The proposed amendment would clarify the power level threshold at which certain reactor protective system (RPS) instrumentation trips must be enabled and may be bypassed, and clarify that this level is a percentage of the neutron flux at rated thermal power (RTP). The bypass power level, 1E–4% RTP, would be specified as logarithmic power instead of thermal power. The intent of (and the implementation of) the 1E–4% RTP RPS instrumentation bypass threshold level in the technical specifications (TS) has always been that this power level is neutron power, which would be indicated by logarithmic power, and is not the heat transfer from the reactor core to the coolant, including decay heat, which is the thermal power definition in 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would replace the words ''THERMAL POWER'' with ''logarithmic power'' for the 1E–4% rated thermal power (RTP) level threshold in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnote (d) for the reactor protective system (RPS) instrumentation. The purpose of the 1E–4% RTP threshold is to (1) specify the power, below which, the logarithmic power level trip is required to be operable and surveilled, and (2) specify the power, above which, the local power density (LPD) and departure from nucleate boiling ratio (DNBR) trips are required to be operable. For these purposes, the appropriate power threshold should be logarithmic power, which is the power indicated on the logarithmic nuclear instrumentation, and not thermal power. Thermal power is defined in TS section 1.1 as the total reactor heat transfer rate to the reactor coolant, and would include decay heat. Thermal power would therefore not drop to 1E–4% RTP for a considerable period of time after shutdown, and would not provide the plant protective function correlation required at 1E–4% neutron RTP. However, logarithmic power, which is indicated by neutron flux, does provide the plant protective function correlation required at 1E–4% neutron RTP for the required reactor trips as required by safety analyses. The logarithmic power level of 1E–4% neutron RTP nominally correlates to the neutron flux measured by the excore neutron instrumentation that is 1E–4% of the neutron flux at 100% RTP (3876 MWt) measured by the excore neutron instrumentation.

The proposed editorial amendment would also replace ''RTP'' with ''NRTP,'' in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnotes (c) and (d). A definition would be added for NRTP (nuclear rated thermal power) in section 1.1 as the indicated neutron flux at RTP. These editorial clarifications will reflect the fact that the logarithmic power level of 1E–4% is not a percentage of the ''total reactor core heat transfer rate to the reactor coolant of 3876 MWt,'' as RTP is defined in section TS 1.1, but is instead a percentage of the indicated neutron flux at RTP.

An editorial change is also proposed to specify NRTP as the ''ALLOWABLE VALUE'' parameter for the high logarithmic power level trip setpoint in Table 3.3.1–1 to correct the unintended omission of the trip setpoint parameter during preparation of the Improved Technical Specifications. This change will fill in the omitted parameter with the correct parameter of NRTP that is also consistent with the high logarithmic power trip setpoint parameter in Table 3.3.2–1.

These changes do not constitute a physical change to the Unit or make changes in the RPS instrumentation setpoints, system logic or manual actuation. In addition, these changes do not alter physical plant equipment or the way in which plant equipment is operated. This change is editorial in that it corrects the TS wording to match the appropriate power parameter that was originally intended and required by safety analyses, and that has been implemented since original licensing of the PVNGS plants. Therefore, these 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 change would replace the words ''THERMAL POWER'' with ''logarithmic power'' for the 1E–4% RTP level threshold in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnote (d) for the RPS instrumentation. The purpose of the 1E–4% RTP threshold is to (1) specify the power, below which, the logarithmic power level trip is required to be operable and surveilled, and (2) specify the power, above which, the LPD and DNBR trips are required to be operable. For these purposes, the appropriate power threshold should be logarithmic power, which is the power indicated on the logarithmic nuclear instrumentation, and not thermal power. Thermal power is defined in TS section 1.1 as the total reactor heat transfer rate to the reactor coolant, and would include decay heat. Thermal power would therefore not drop to 1E–4% RTP for a considerable period of time after shutdown, and would not provide the plant protective function correlation required at 1E–4% neutron RTP. However, logarithmic power, which is indicated by neutron flux, does provide the plant protective function correlation required at 1E–4% neutron RTP for the required reactor trips as required by safety analyses.

The proposed editorial amendment would also replace ''RTP'' with ''NRTP,'' in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnotes (c) and (d). A definition would be added for NRTP (nuclear rated thermal power) in section 1.1 as the indicated neutron flux at RTP. These editorial clarifications will reflect the fact that the logarithmic power level of 1E–4% is not a percentage of the ''total reactor core heat transfer rate to the reactor coolant of 3876 MWt,'' as RTP is defined in section TS 1.1, but is instead a percentage of the indicated neutron flux at RTP.

An editorial change is also proposed to specify NRTP as the ''ALLOWABLE VALUE'' parameter for the high logarithmic power level trip setpoint in Table 3.3.1–1 to correct the unintended omission of the trip setpoint parameter during preparation of the Improved Technical Specifications. This change will fill in the omitted parameter with the correct parameter of NRTP that is also consistent with the high logarithmic power trip setpoint parameter in Table 3.3.2–1.

These changes do not constitute a physical change to the Unit or make changes in the RPS instrumentation setpoints, system logic or manual actuation. In addition, these changes do not alter physical plant equipment or the way in which plant equipment is operated. The proposed change does not introduce any new modes of plant operation or new accident precursors. This change is editorial in that it corrects the TS wording to match the appropriate power parameter that was originally intended and required by safety analyses, and that has been implemented since original licensing of the PVNGS plants. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would replace the words ''THERMAL POWER'' with ''logarithmic power'' for the 1E–4% RTP level threshold in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnote (d) for the RPS instrumentation. The purpose of the 1E–4% RTP threshold is to (1) specify the power, below which, the logarithmic power level trip is required to be operable and surveilled, and (2) specify the power, above which, the LPD and DNBR trips are required to be operable. For these purposes, the appropriate power threshold should be logarithmic power, which is the power indicated on the logarithmic nuclear instrumentation, and not thermal power. Thermal power is defined in TS section 1.1 as the total reactor heat transfer rate to the reactor coolant, and would include decay heat. Thermal power would therefore not drop to 1E–4% RTP for a considerable period of time after shutdown, and would not provide the plant protective function correlation required at 1E–4% neutron RTP. However, logarithmic power, which is indicated by neutron flux, does provide the plant protective function correlation required at 1E–4% neutron RTP for the required reactor trips as required by safety analyses.

The proposed editorial amendment would also replace ''RTP'' with ''NRTP,'' in Table 3.3.1–1 footnotes (a) and (b), surveillance requirement SR 3.3.1.7 Note 2, and Table 3.3.2–1 footnotes (c) and (d). A definition would be added for NRTP (nuclear rated thermal power) in section 1.1 as the indicated neutron flux at RTP. These editorial clarifications will reflect the fact that the logarithmic power level of 1E–4% is not a percentage of the ''total reactor core heat transfer rate to the reactor coolant of 3876 MWt,'' as RTP is defined in section TS 1.1, but is instead a percentage of the indicated neutron flux at RTP.

An editorial change is also proposed to specify NRTP as the ''ALLOWABLE VALUE'' parameter for the high logarithmic power level trip setpoint in Table 3.3.1–1 to correct the unintended omission of the trip setpoint parameter during preparation of the Improved Technical Specifications. This change will fill in the omitted parameter with the correct parameter of NRTP that is also consistent with the high logarithmic power trip setpoint parameter in Table 3.3.2–1.

These changes do not constitute a physical change to the Unit or make changes in the RPS instrumentation setpoints, system logic or manual actuation. In addition, these changes do not alter physical plant equipment or the way in which plant equipment is operated. This change is editorial in that it corrects the TS wording to match the appropriate power parameter that was originally intended and required by safety analyses, and that has been implemented since original licensing of the PVNGS plants. 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 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: 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.

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

Date of amendment request: October 14, 1998.

Description of amendment request: The proposed change will revise the H. B. Robinson, Unit 2, Technical Specification (TS) on Residual Heat Removal Isolation Valve Interlock. The requested change modifies the acceptance criterion for surveillance requirement (SR) 3.4.14.2 from setpoint value to the analytical limit for overpressurization of the Residual Heat Removal System.

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

The HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2 TS are proposed to be modified to increase the acceptance criterion for Surveillance Requirement (SR) 3.4.14.2 from a RCS [reactor coolant system] pressure of 465 psig to 474 psig. Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specifications (TS) change and has concluded that it does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration is discussed below.

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

The proposed change increases the acceptance criterion for the Residual Heat Removal (RHR) System interlock from 465 psig to 474 psig. The new value of 474 psig is the analytical limit for the RHR System interlock setpoint that corresponds to the highest RCS pressure that is allowable in the RHR System without overpressurizing the RHR System above its design pressure. The RHR System interlock prohibits remote manual operation of the RHR Pressure Isolation Valves (PIVS) from the control room when Reactor Coolant System (RCS) pressure is greater than the RHR System interlock setpoint to avoid inadvertent overpressurization of the RHR System due to operator action. Operating procedures prohibit opening of the RHR PIVs when RCS pressure is greater than 375 psig. Therefore, the probability of overpressurization of the RHR System resulting in a Loss-of-Coolant Accident (LOCA) is not affected by the change. The RHR System interlock provides no actuation function to mitigate the consequences of a LOCA as a result of open RHR PIVs with RCS pressure greater than the RHR System interlock setpoint. Therefore, the consequences of overpressurization of the RHR System is not affected by the change. Therefore, the proposed change does not involve any 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 change does not involve any physical alteration of plant systems, structures, or components. The proposed change increases the acceptance criterion for the RHR System interlock SR from 465 psig to the analytical limit of 474 psig. Performance of a SR at the new acceptance criterion does not introduce any new accident initiation scenarios since the SR is performed at acceptable RCS pressure conditions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change results in a new SR acceptance criterion that corresponds to the analytical limit for the RHR System interlock setpoint. The RHR System interlock is redundant to administrative controls which prohibit opening the RHR System PIVs under RCS pressure conditions which would overpressurize the RCS System. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

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

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

Date of application for amendment request: October 13, 1998.

Description of amendment request: The proposed amendments would change the Dresden, Quad Cities, and LaSalle Technical Specifications (TS) to reflect the use of Siemens Power Corporation (SPC) ATRIUM–9B fuel. Specifically the proposed amendments incorporate the following into the TS: (a) new methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates; (b) administrative changes that eliminate the cycle-specific implementation of ATRIUM–9B fuel and adopt Improved Standard Technical Specification language where appropriate; and (c) changes to the Minimum Critical Power Ratio (MCPR). This amendment request supplements the submittal of August 14, 1998 (63 FR 48258). Changes in this supplement include only a change in reference to a recently NRC-approved additive constant uncertainty (ACU) generic methodology for ATRIUM–9B fuel (ANF–1125(P)(A), Supplement 1, Appendix E) from Appendix D which provided an interim value for ACU.

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The Reference 1 [ANF–91–048(P)(A), Supplement 1 and Supplement 2, ''BWR Jet Pump Model Revision for RELAX,'' October 1997 and NRC SER, ''Review of Siemens Topical Report ANF–91–048(P), BWR Jet Pump Revisison for RELAX (TAC No M995381), T.H. Essig to H.D. Curet, September 19, 1997] methodology to be added to the Technical Specifications is used as part of the LOCA [loss-of-coolant accident] analysis and does not introduce physical changes to the plant. The Reference 1 revised jet pump model changes the calculational behavior of the jet pump under reversed drive flow conditions. The revised jet pump model methodology makes the LOCA model behave more realistically and calculates small break LOCA PCTs [peak cladding temperature] that are comparable to the large break LOCA results. Therefore, this change only affects the methodology for analyzing the LOCA event and determining the protective APLHGR [average planar linear heat generation rate] limits. The Technical Specification requirements for monitoring APLHGR are not affected by this change. The revised method will result in higher APLHGR limits, thus the SPC fuel will be allowed to operate at higher nodal powers. The approved methodology, however, still protects the fuel performance limits specified by 10 CFR 50.46. Therefore, the probability or consequences of an accident previously evaluated will not change.

b. Addition of SPC Generic Methodology for Application of ANFB [Advanced Nuclear Fuel for Boiling Water Reactors] Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 3 [EMF–1125(P)(A), Supplement 1 Appendix C, ''ANFB Critical Power Correlation Application for Coresident Fuel,'' August 1997, and NRC SER, ''Acceptance for Referencing of Licensing Topical Report EMF–1125(P), Supplement 1 Appendix C, ''ANFB Critical Power Correlation Application for Co-Resident Fuel,'' J.E. Lyons to R. A. Copeland, May 9, 1997] to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 3 determines the additive constants and the associated uncertainty for application of the ANFB correlation to the coresident GE [General Electric Co.] fuel. Therefore, it provides data that is used in the determination of the MCPR Safety Limit. This approved methodology for applying the ANFB critical power correlation to the GE fuel will protect the fuel from boiling transition. Operational MCPR limits will also be applied to ensure that the MCPR Safety Limit is protected during all modes of operation and anticipated operational occurrences. Because Reference 3 contains conservative methods and calculations and because the operability of plant systems designed to mitigate any consequences of accidents have not changed, the probability or consequences of an accident previously evaluated will not increase.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 7 [ANF–1125(P), Supplement 1, Appendix E, ''ANFB Critical Power Correlation Determination of ATRIUM–9B Additive Constant Uncertainties,'' and NRC SER, ''Acceptance for Referencing of Licensing Topical Report ANF–1125(P), Supplement 1, Appendix E, ''ANFB Critical Power Correlation Determination of ATRIUM–9B Additive Constant Uncertainties'' (TAC No. MA2437), T.H. Essig to H.D. Curet, September 23, 1998] to Section'' 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 7 documents the additive constant uncertainty for the SPC ATRIUM–9B fuel design with an internal water channel. This methodology is used to determine an input to the MCPR Safety Limit calculations, which ensures that at least 99.9 percent of the fuel rods avoid transition boiling during normal operation as well as anticipated operational occurrences. This change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. This methodology for determining the ATRIUM–9B additive constant uncertainty for the MCPR Safety Limit calculation will continue to support protecting the fuel from boiling transition. Operational MCPR limits will be applied to ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Therefore, no individual precursors of an accident are affected and the operability of plant systems designed to mitigate the probability or the consequences of an accident previously evaluated is not affected by these changes.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit at Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 will not increase the probability or the consequences of an accident previously evaluated. The MCPR Safety Limits for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 are anticipated to be conservative and acceptable for future cycles. Cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. The MCPR Safety Limits are being set at the CPR [critical power ratio] value where less than 0.1 percent of the rods in the core are expected to experience boiling transition. These Safety Limits are expected to be applicable for future cycles of ATRIUM–9B. Therefore the probability or consequences of an accident will not increase.

e. Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of footnotes from the Quad Cities and Dresden Technical Specifications does not involve any significant increase in the probability or consequences of an accident previously evaluated. The footnotes were added to clarify that cycle specific methods were used until the generic methodology was approved by the NRC. Since the NRC has approved SPC's generic methodology for application of the ANFB correlation to the coresident GE fuel (Reference 3) and SPC has addressed the concerns regarding the database used to calculate the ATRIUM–9B additive constant uncertainties (Reference 7), the footnotes are no longer necessary. The removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, removing these footnotes and ''a'' pages does not require any physical plant modifications, nor does it physically affect any plant components or entail changes in plant operation. Therefore, the probability or consequences of an accident previously evaluated are not expected to increase.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the Section 3 Technical Specification description of the APLHGR limits has no implications on accident analysis or plant operations. The purpose of the revision is to allow flexibility for the MAPLHGR [maximum planar linear heat generation rate] limits and their exposure basis to be specified in the COLR [core operating limit report] and to establish consistency with approved methodologies currently utilized by Siemens Power Corporation, which calculate MAPLHGR limits based on bundle or planar average exposures. This revision also provides for consistency in the APLHGR limit Technical Specification wording between the ComEd BWRs. The revision to the 3.11.D SLHGR [steady state linear heat generation rate] Technical Specification for Dresden also has no implications on accident analysis or plant operations. The purpose of this revision is to allow flexibility for the LHGR [linear heat generation rate] limits and their exposure basis to be specified in the COLR. This revision makes the Dresden LHGR definition consistent with NUREG 1433/1434, Revision 1 wording. The definition of the Average Planar Exposure is deleted, because the exposure basis of the APLHGR and LHGR is being removed. Therefore, no plant equipment or processes are affected by this change. Thus, there is no alteration 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:

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications to the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications to the plant configuration or allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology will be used to analyze the LOCA for LaSalle Units 1 and 2, and does not introduce any physical changes to the plant or the processes used to operate the plant. This change only affects the methods used to analyze the LOCA event and determine the MAPLHGR limits. Therefore, the possibility of a new or different kind of accident is not created.

b. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Addition of the generic methodology for the application of the ANFB critical power correlation to GE fuel in Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. This change only involves adding an NRC approved methodology, which is used to determine the additive constants and additive constant uncertainty for GE fuel, to Section 6 of the Technical Specifications. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Addition of the Reference 7 methodology to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications will not create the possibility of a new or different kind of accident from any accident previously evaluated. This methodology describes the calculation of an input to the MCPR Safety Limit—the ATRIUM–9B additive constant uncertainty. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit will not create the possibility of a new accident from an accident previously evaluated. This change will not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9 percent of the rods avoid boiling transition.

The MCPR Safety Limit is changing for Quad Cities, Dresden Unit 3 and LaSalle due to the revised ATRIUM–9B additive constants and the ATRIUM–9B additive constant uncertainty calculated in Reference 7. The new MCPR Safety Limit for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 are greater than the current values at Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 and are being increased now in anticipation of bounding future reloads of ATRIUM–9B. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

e. Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of the footnotes from the Quad Cities and Dresden Technical Specifications does not create a new or different kind of accident from any accident previously evaluated. The removal of the footnotes does not affect plant systems or operation. The footnotes were temporarily established to implement a conservative cycle specific MCPR Safety Limit until the SPC generic methodology was approved. With the approval of References 3 and 7, these footnotes are no longer applicable. Removing these footnotes does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. The removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications, which is justified by the removal of the footnotes, also does not create a new or different kind of accident from any accident previously evaluated.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle 1 and 2)

The revision of the APLHGR and LHGR limit descriptions will not create the possibility of a new or different kind of accident from any accident previously evaluated. This revision will not alter any plant systems, equipment, or physical conditions of the site. This revision allows the flexibility of the APLHGR and the LHGR limits to be specified in the COLR and to maintain consistency with the calculated results of methodologies currently used to determine the APLHGR. The definition of the Average Planar Exposure is deleted, because it is being removed from LHGR and APLHGR Technical Specifications. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology, and the MAPLHGRs, resulting from the revised jet pump methodology, will continue to ensure fuel design criteria and 10 CFR 50.46 compliance. The results of LOCA analyses performed with this methodology must continue to comply with the requirements of 10 CFR 50.46. Therefore, there is no significant reduction in the margin of safety.

b. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The margin of safety is not decreased by adding Reference 3 to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Siemens Power Corporation methodology for application of the ANFB Critical Power Correlation to coresident GE fuel is approved by the NRC and is the same methodology used in the cycle specific topicals for coresident fuel (References 4 [EMF–96–021(P), Revision 1, Application of the ANFB Critical Power Correlation to Coresident GE fuel for LaSalle Unit 2 Cycle 8,'' February 1996, and NRC SER, ''Safety Evaluation for Topical Report EMF–96–021(P), Revision 1, 'Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8' (TAC NO. M94964),'' D.M. Skay to I. Johnson, September 26, 1996] and 5 [EMF–96–051(P), ''Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15,'' May 1996, and NRC SER, ''Approval of Topical Report EMF–96–051(P)—Quad Cities, Unit 2 (TAC NO. M96213),'' R. Pulsifer to I. Johnson, May 16, 1997]). The MCPR Safety Limit will continue to ensure that greater than 99.9 percent of the rods in the core avoid boiling transition. Additionally, operating limits will be established to ensure the MCPR Safety Limit is not violated during all modes of operation.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1 percent of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. This Technical Specification amendment request proposes to insert the topical report that describes SPC's calculation of the ATRIUM–9B additive constant uncertainty. The new ATRIUM–9B additive constant uncertainty calculation is conservative and is based on a larger database than previous calculations. Because the criteria of ensuring that 99.9 percent of the rods are expected to avoid boiling transition has not been changed and a conservative method is used to calculate the ATRIUM–9B additive constant uncertainty, a decrease in the margin to safety will not occur due to adding this methodology to the Technical Specifications. In addition, operational limits will be established to ensure the MCPR Safety Limit is protected for all modes of operation. This revised methodology will ensure that the appropriate level of fuel protection is being employed.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 will not involve any reduction in margin of safety. The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1 percent of the rods are calculated to be in boiling transition if the MCPR Safety Limit is not violated. The proposed Technical Specification amendment request reflects the MCPR Safety Limit results from conservative evaluations by SPC using the ANFB critical power correlation with the ATRIUM–9B additive constant uncertainty calculated in Reference 7.

Because a conservative method is used to apply the ATRIUM–9B additive constant uncertainty in the MCPR Safety Limit calculation, a decrease in the margin to safety will not occur due to changing the MCPR Safety Limit. The revised MCPR Safety Limit will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR Safety Limit to ensure that the MCPR Safety Limit is not violated during all modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9 percent of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

e. Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of the cycle specific footnotes in Quad Cities and Dresden Technical Specifications does not impose a change in the margin of safety. These footnotes were added due to concerns regarding the calculation of the additive constant uncertainty for the ATRIUM–9B fuel and the cycle specific application of the ANFB critical power correlation to coresident GE fuel in Quad Cities Unit 2 Cycle 15. Because the generic ANFB application to coresident GE fuel MCPR methodology (Reference 3) has received NRC approval and the topical report describing the increased database used to calculate the additive constant uncertainties for ATRIUM–9B (Reference 7) has also received NRC approval and both are proposed to be added to the Technical Specifications in this amendment request, there is no reason for the footnotes to remain. Removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, the removal of the ''a'' pages, 2–1a and B2–3a, also does not impose a change in the margin of safety.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the APLHGR and LHGR limit descriptions will not involve a reduction in the margin of safety. The methodology used to calculate the APLHGR must comply with the guidelines of Appendix K of 10 CFR Part 50, and the APLHGR and LHGR will still be required to be maintained within the limits specified in the COLR. The surveillance requirements for these two thermal limits remain unchanged. Thus, there will be no 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: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021; and for LaSalle, the Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603. *NRC Project Director:* Stuart A. Richards.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: September 30, 1998.

Description of amendment request: The proposed amendment would increase the maximum fuel rod internal pressure in the spent fuel pool from 1200 pounds per square inch gauge (psig) to 1300 psig by changing the Updated Final Analysis Report (UFSAR) reference to the computer code used to determine the fuel rod internal pressure (TACO3 computer code would be added) in UFSAR Chapter 15. The proposed amendment would also provide justification for not increasing the overall effective decontamination factor for iodine as a consequence of a fuel handling accident. In addition, the term ''fuel assembly gap gas pressure'' would be changed to ''fuel rod internal pressure'' to correct an UFSAR error.

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

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92 (c) requirements to demonstrate that all three standards for no significant hazards consideration are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

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

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

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

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The increase in maximum rod internal pressure in the spent fuel pool from 1200 psig to 1300 psig does not result in a significant change in the calculated overall effective decontamination factor for iodine (described in Attachment 1) [of the licensee's submittal]. Therefore, the continued use of an overall effective decontamination factor for iodine of 89 can be justified. Therefore, there is no significant increase in the dose consequences for a fuel handling accident at Oconee Nuclear Station.

Implementation of the BAW–10183P-A (Reference 4) methodology, which allows fuel rod internal pressure to exceed system pressure, also increases the fuel rod pressure at spent fuel pool conditions. The fuel is currently licensed to rod internal pressure of system pressure plus a proprietary amount above system pressure. This criteria represents a separate limit from the maximum internal pressure in the spent fuel pool criteria. Thus, an increase in the maximum rod internal pressure in the spent fuel pool does not affect the mechanical design limit specified in Reference 4. Therefore, an increase in the maximum internal pressure in the spent fuel pool does not constitute a significant increase in the probability of an accident previously evaluated.

Second Standard

Implementation of this amendment will not create the possibility of a new or different kind of accident from any previously evaluated. The fuel handling accident is the bounding accident. Implementation of this amendment will not impact any plant systems that are accident initiators. No other modifications are being proposed in the plant which would result in the creation of a new accident mechanism. Also, no changes are being made to the way the plant is operated; therefore, no new failure mechanisms will be initiated.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. As discussed in Attachment 1 [of the licensee's submittal], the overall effective decontamination factor (DF) of 522 was determined for a rod internal pressure of 1200 psig, and a DF of 443 for a rod internal pressure of 1300 psig based on a spent fuel pool depth of 21.34 feet. Both of these factors are well above the DF of 89 currently used in the fuel handling accident analyses. The margin of safety is a factor of 5.

Based upon the preceding analysis, Duke proposes that ample margin is retained to justify the continued use of a DF of 89 at a maximum rod internal pressure of 1300 psig. Therefore, Duke has concluded that the proposed amendment does not involve a significant hazards consideration.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

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

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: September 24, 1998.

Description of amendment request: The proposed amendment would revise technical specification (TS) 3.1.2.8 in two places to change the term ''contained volume'' to ''usable volume.'' This change would eliminate the potential for a non-conservative interpretation of the specification values for the Refueling Water Storage Tank and Boric Acid Storage System (BAT) and would eliminate the need for plant administrative controls, which currently interpret these volumes as usable volumes.

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 Limiting Condition for Operation (LCO) change will assure that the Refueling Water Storage Tank (RWST) minimum usable volume is maintained consistent with that required by accident analysis. The safety function of the RWST will not differ in any way from its normal operational mode. The normal operation of plant equipment is not a precursor to any accident. Therefore, operation of equipment under this change will not increase 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 amendment will not change the physical plant or the modes of plant operation defined in the operating license. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. The proposed change will help to ensure that the analysis value of minimum contained volume is available, so that the RWST can perform its safety function.

Therefore, operation of the facility in accordance with the proposed amendment will 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?

RWST: The basis for TS 3.1.2.8.b is to ensure adequate water for the Emergency Core Cooling System to respond to a Large Break Loss Of Coolant Accident; supply the containment with cooling spray flow; supply the containment sump with adequate water for Recirculation Spray pump suction head concerns; and to provide adequate boron to shut down the core. This change will ensure that the proper tank volume is maintained to support the Design Basis Accident (DBA) analysis.

BAT: These tanks are credited for ensuring adequate Shutdown Margin in the event that the unit has to initiate an emergency shutdown. Additional requirements are derived for the postulated Anticipated Transient Without Scram event. This change will ensure that the proper tank volume is maintained to support the DBA analysis.

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, PA 1500l.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: October 16, 1998.

Description of amendment request: The proposed amendment would extend on a one time only basis, the surveillance interval for technical specifications (TSs) 4.8.1.1.1.b and 4.8.1.2 from its current due date of January 30, 1999, to the first entry into Mode 4 following the seventh refueling outage (2R7), but not later than May 1, 1999, by adding a new License Condition 2.C(12). The purpose of TSs 4.8.1.1.1.b and 4.8.1.2 is to demonstrate the ability to transfer the unit power supply from the unit circuit to the system circuit.

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 change is temporary and allows a one time extension of the automatic transfer function 18 month surveillance requirement specified in Surveillance Requirement (SR) 4.8.1.1.1.b. This surveillance requirement is also referenced in SR 4.8.1.2. The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of a system to perform its design function. The proposed change does not affect the UFSAR [Updated Final Safety Analysis Report] accident analyses since a loss of offsite power is assumed during a design basis accident. Therefore, this change does 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?

Extending the surveillance interval for the performance of specific testing will not create the possibility of any new or different kind of accidents. No change is required to any system configurations, plant equipment or analyses. The UFSAR accident analyses assume a loss of offsite power; therefore, loss of the automatic bus transfer feature will 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?

Extending the surveillance interval for the automatic transfer function will not impact any plant safety analyses since the UFSAR accident analyses assume the loss of offsite power. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the 18 month surveillance test interval is being extended. Based on engineering judgment, extending the surveillance test interval for the performance of this specific test could slightly reduce the margin of safety derived from the required surveillances. However, past experience has shown that the system which automatically transfers power from the unit to the system circuit supply is reliable. The manual transfer requirement of SR 4.8.1.1.1.b demonstrates that the breakers relied upon for the transfer of power are functional and provides an opportunity to identify potential equipment degradation. The manual transfer requirement of SR 4.8.1.1.1.b will continue to be completed within the required surveillance interval. Therefore, the plant will be maintained within the analyzed limits and the proposed extension will not significantly reduce 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, PA 1500l.

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

NRC Project Director: Robert A. Capra.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: September 22, 1998.

Description of amendment request: The proposed amendment would delete license conditions associated with the River Bend Station (RBS) Transamerica Delaval, Inc. (TDI) emergency diesel generators (EDGs), which prescribe certain inspection requirements associated with various overload conditions experienced by the EDGs. Current license requirements were issued following publication of NUREG–1216, which called for extensive periodic engine tear-downs as the major part of a maintenance and surveillance program for TDI engines. The proposed removal of license conditions appears to be consistent with the NRC's approval of Generic Topical Report TDI–EDG–001–A ''Basis for Modification to Inspection Requirements for Transamerica Delaval, Inc., Emergency Diesel Generators''. EOI currently inspects and maintains its EDGs in accordance with Technical Requirements Manual (TRM) surveillance requirement TSR 3.8.1.21. Periodicity of planned inspections and maintenance are based upon the manufacturer's recommendations for standby service.

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 the consequences of an accident previously evaluated:

Diesel generators are not accident initiating equipment. Elimination of the non-routine tear-downs and inspections will not adversely affect the probability of an accident occurring. Regular maintenance programs (which may include periodic tear-downs and inspections) in lieu of this specific license condition would decrease the consequences of an accident because of the availability of the engines will increase as a result of the less frequent tear-downs. (See Generic Topical Report TDI–EDG–001–A, ''Basis for Modification to Inspection Requirements for Transamerica Delaval, Inc., Emergency Diesel Generators'') Additionally, the high average reliability of the TDI engines will not be negatively affected due to this change. NRC research has shown there is a period of decreased reliability immediately following intrusive tear-downs (break-in period), followed by a long period of high reliability. Continued monitoring and maintenance as implemented by Technical Requirements Manual (TRM) surveillances will contribute to continued high reliability of the EDGs.

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

The proposed amendment does not affect the design or function of any plant structure, system, or component, nor does it change the way plant systems are operated. The proposed amendment will not cause any physical change to the plant or the design or operation of the diesel units. This change will only affect the frequency of tear-down inspections of the EDGs, and not the physical activities performed during such inspections. Therefore, the removal of the existing condition from the operating license will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment does not affect parameters which would result in a significant reduction in margin of safety. Operating experience and data have shown increased reliability can be achieved by eliminating unnecessary tear-down inspections, such as those prescribed by this license condition. Maintenance of the EDGs is presently scheduled in accordance with the vendor's recommendations. The RBS corrective action program provides a means to evaluate future operational events and take the appropriate actions. Therefore, the proposed amendment does not involve a significant decrease 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005.

NRC Project Director: John N. Hannon.

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

Date of amendment request: May 28, 1998.

Description of amendment request: This amendment requests changes to Technical Specification 3.7.1.2 and Surveillance Requirement 4.7.1.2 for the Emergency Feedwater System. The amendment will expand and clarify the current specification. A change to Technical Specification Bases 3/4.7.1.2 has been included to support the changes.

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

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

Response: No.

The proposed changes included in this amendment request are being made to the Emergency Feedwater (EFW) System Technical Specification. These changes include clarification of the LCO [limiting conditions for operation], a 7 day allowed outage time for an inoperable steam supply, additional ACTION requirements for inoperable flow path(s), a requirement to test the pumps pursuant to Specification 4.0.5, and rewording of numerous Surveillance Requirements consistent with NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants.''

The administrative and more restrictive changes will not affect the assumptions, design parameters, or results of any accident previously evaluated. The accident mitigation features of the plant are not affected by these proposed changes. The proposed changes do not add or modify any existing equipment. The administrative change to test EFW pumps pursuant to the Inservice Test Program will ensure the EFW pumps are tested against the more restrictive of the data points required by either the safety analysis or the Inservice Test Program. Therefore, the proposed administrative changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

The less restrictive changes (allowing 7 days for an inoperable pump due to an inoperable steam supply, performing Surveillance Requirements during other than shut down conditions, allowing the use of actual actuation signals in addition to test signals, and delaying the requirement to complete Surveillance Requirement ''d'' to just prior to Mode 2) will not affect the assumptions, design parameters, or results of any accident previously evaluated. The accident mitigation features of the plant are not affected by these proposed changes. The proposed changes do not add or modify any existing equipment. Therefore, the proposed less restrictive changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

The proposed changes included in this amendment request are being made to the EFW System Technical Specification. These changes include clarification of the LCO, a 7 day allowed outage time for an inoperable steam supply, additional ACTION requirements for inoperable flow path(s), a requirement to test the pumps pursuant to Specification 4.0.5, and rewording of numerous Surveillance Requirements consistent with NUREG–1432. These changes do not alter the design nor configuration of the plant. There has been no physical change to plant systems, structures, or components. The proposed changes will not reduce the ability of any of the safety-related equipment required to mitigate Anticipated Operational Occurrences or accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes included in this amendment request are being made to the EFW System Technical Specification. These changes include clarification of the LCO, a 7 day allowed outage time for an inoperable steam supply, additional ACTION requirements for inoperable flow path(s), a requirement to test the pumps pursuant to Specification 4.0.5, and rewording of numerous Surveillance Requirements consistent with NUREG–1432.

The proposed change to the LCO requiring three pumps and two flow paths be OPERABLE maintains the functionality of the EFW such that it is capable of performing its design function as assumed in the Updated Final Safety Analysis Report. If the functionality of the system is not maintained, Technical Specifications require ACTIONs be taken, within specified time limitations, to restore EFW to OPERABLE status or shut down the reactor. This action is consistent with the existing Technical Specification and NUREG–1432.

The allowed outage time for one inoperable steam supply has been increased from 72 hours to 7 days in accordance with NUREG–1432. This is acceptable due to the redundant OPERABLE steam supply, the availability of redundant OPERABLE motor-driven EFW pumps, and the low probability of an event requiring the inoperable steam supply. This change is consistent (other than format) with NUREG–1432 and has therefore been previously approved by the NRC.

The ACTION for one flow path inoperable (but capable of delivering 100% flow) as proposed will allow a 72 hour completion time for an inoperable flow path. This change is acceptable based on the availability of at least two OPERABLE EFW pumps, a redundant OPERABLE flow path capable of feeding the other steam generator and the capability of the inoperable flow path to deliver 100% of the required EFW flow to the affected steam generator.

The ACTION for one flow path inoperable (not capable of delivering 100% flow) as proposed requires a unit shutdown be initiated immediately. This change is appropriate due to the seriousness of the condition and is acceptable due to the availability of the remaining operable flow path to support the unit shut down.

The ACTION for two flow paths not capable of delivering 100% flow is the same as that for three pumps inoperable. With two flow paths inoperable such that neither flow path is capable of delivering 100% flow the unit is in a seriously degraded condition just as it is with all three pumps inoperable. The ACTION as proposed requires that immediate action be taken to restore one flow path to OPERABLE status. This change is consistent with the intent of the current EFW Technical Specification.

Testing pursuant to Specification 4.0.5 (Inservice Testing Program) as proposed for Surveillance Requirement 'b' will ensure the EFW pumps are tested against the more restrictive of the data points required by either the safety analysis or ASME Section XI.

The remaining changes to the EFW Technical Specification are consistent (other than format) with NUREG–1432 and have therefore been previously approved by the NRC.

Therefore, based on the above discussion, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: John N. Hannon.

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

Date of amendment request: September 28, 1998.

Description of amendment request: In 1997 Northeast Nuclear Energy Company (the licensee) changed the Final Safety Analysis Report (FSAR) Section 8.7.3.1 electrical separation requirements from 12 inches to 6 inches. At that time, the licensee concluded that the FSAR changes did not involve an unreviewed safety question. Therefore, the licensee did not request a license amendment to implement the FSAR change. The licensee has since determined that, although the changes were safe, an unreviewed safety question was involved. Therefore, the licensee is now requesting NRC's review and approval, through an amendment to Operating License No. DPR–65 pursuant to 10 CFR 50.90, regarding the separation requirement of 6 inches in Millstone Unit No. 2 FSAR (which is applied to redundant vital cables, internal wiring of redundant vital circuits, and associated devices).

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:

In accordance with 10CFR50.92, NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes and has concluded that they do 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 changes do not involve an SHC because the changes would not:

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

The FSAR changes reduce the minimum allowable separation between redundant vital wires/devices of different channels from twelve inches to six inches. Reducing the physical separation between wires/devices does not in itself increase the probability of any credible event that would challenge circuit operability since the wire/device characteristics have not changed and there is no change in the circuit the wires/devices are in. The probability that an accident could occur due to the change in separation is not increased since the remaining separation will still prevent adverse channel interactions (i.e. short circuit, etc.). The six inch standard is acceptable in accordance with IEEE standard 384–1981 [IEEE standard 384–1981, ''Standard Criteria for Independence of Class 1E Equipment and Circuits''], sections 6.6.2 and 6.6.5, and IEEE standard 420–1982, [IEEE standard 420–1982, ''Design Standards and Qualification of class 1E Control Boards, panels, and Racks Used in Nuclear Power Generating Stations''], sections 4.3.1, 4.3.2, and 4.3.3 which have been endorsed by the NRC in Regulatory Guide 1.75 [Regulatory Guide 1.75, ''Physical Independence of Electrical Systems'']. Therefore, these changes will not significantly increase 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 FSAR changes reduce the minimum allowable separation between redundant vital wires/devices of different channels from twelve inches to six inches. The new minimum allowable separation will not introduce any new or unanalyzed failure modes of equipment or systems, and does not change the configuration of the plant. These changes will not require any new or unusual operator actions, alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. Therefore, there are no new or different types of failures of systems or equipment important to safety which could cause a new or different type of accident from any accident previously evaluated.

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

The FSAR changes reduce the minimum allowable separation between redundant vital wires/devices of different channels from twelve inches to six inches. The probability that a single wire/device failure could cause the failure of redundant vital channels may be increased. However, the new minimum allowed separation has been found acceptable by IEEE standard 384–1981, sections 6.6.2 and 6.6.5, and IEEE standard 420–1982, sections 4.3.1, 4.3.2, and 4.3.3 which have been endorsed by the NRC in Regulatory Guide 1.75. The new minimum allowed separation does not change any plant equipment configuration, does not change the functionality of any equipment, and does not change any operating setpoints. This change does not alter the acceptance limits of the safety parameters of the accident analyses stated in the FSAR. No new analysis assumptions are required based on this change (e.g. common-cause failures). Therefore, there is no impact on 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, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Project Director: William M. Dean.

Public Service Electric & Gas Company, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: October 12, 1998.

Description of amendment request: The proposed amendment would allow a one-time extension of the Technical Specification (TS) surveillance interval to the end of fuel cycle 10 for certain TS surveillance requirements (SRs). Specifically, SR 4.3.2.1.3 requires the instrumentation response time testing of each engineered safety features actuation system function at least once per 18 months and SRs 4.8.2.3.2.f and 4.8.2.5.2.d require that the 125 volt DC and the 28 volt DC distribution system batteries, respectively, be capacity service tested at least once per 18 months, during shutdown. Additionally, SR 4.8.2.5.2.c.2 requires that the 125 volt DC battery connections be verified clean, tight, and coated with anti-corrosion material at least once per 18 months. Because of the length of the last outage and delays in restart, the SRs will be overdue prior to reaching the next refueling outage (2R10). The SRs are to be completed during the 2R10 outage, prior to returning the unit to Mode 4 (hot shutdown) upon outage completion.

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:

4.3.2.1.3 (Instrumentation, Engineered Safety Feature Actuation System Instrumentation)

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

The deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated. Thus, the proposal does not increase the probability of an accident previously evaluated.

The SEC [safeguard equipment control] automatic self-test feature, the monthly functional surveillance testing and the positive surveillance testing history provide sufficient assurance of the operability of the system. These features also provide assurance that a degraded condition, if it did occur, would be detected.

Thus, it is reasonable to conclude that this proposal represents no significant increase in the consequences of an accident previously analyzed.

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

Deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated.

Thus, it can be concluded that deferring the surveillance requirement to the refueling outage cannot create the possibility of a different kind of accident from any accident previously evaluated.

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

Deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated. The self-test feature and the monthly functional testing will provide reasonable assurance that the SECs will remain operable during the few weeks of deferral to the refueling outage. Also the ability to detect a degraded condition in the SEC will not be affected during the deferral period.

Therefore, the plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 2 Technical Specifications, on a one-time basis, to defer surveillance requirement 4.3.2.1.3 does not involve a significant reduction in any margin of safety.

4.8.2.3.2.f, (Electrical Power Systems, 125 Volt D.C. Distribution), and 4.8.2.5.2.c.2 and 4.8.2.5.2.d (Electrical Power Systems, 28 Volt D.C. Distribution)

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

Weekly and quarterly testing and performance monitoring by the system manager along with the current condition of the batteries (past test results demonstrating above 100% capacity) provide assurance that battery condition and performance will not deteriorate during the deferral period. Other positive industry experience for similar batteries on 24 month cycles also support this assurance. Therefore, the consequences of a loss of power accident will not be increased due to the deferral of the surveillance requirements.

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Continuing weekly and quarterly testing and performance monitoring along with the current condition of the batteries provides assurance that battery condition and performance will not deteriorate to an unacceptable level during the deferral period and that any degradation that may occur will be detected. Therefore, the plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 2 Technical Specifications, on a one-time basis, to defer surveillance requirements 4.8.2.3.2.f and 4.8.2.5.2.d does not involve 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Project Director: Robert A. Capra.

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 request: May 7, 1998.

Description of amendment request: This change would revise the reference for obtaining the thyroid dose conversion factors used in the definition of Dose Equivalent Iodine 131 (I–131) in Technical Specification (TS) Section 1.1, ''Definitions'' for each plant. Specifically, the reference to ''Table E–7 of Regulatory Guide 1.109, Rev. 1, NRC 1977'' is to be replaced with a reference to the International Commission on Radiological Protection Publication 30 (ICRP–30), Supplement to Part 1, Pages 192–212, Tables titled, ''Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity.''

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, which utilizes International Committee on Radiological Protection (ICRP)–30 methodology for determining dose equivalent Iodine-131, and therefore for evaluating thyroid dose consequences, does not involve any change to the method of operation of any plant equipment, nor does it modify any plant equipment. In addition, utilization of the ICRP–30 Dose Conversion Factors (DCFs) will effectively reduce calculated thyroid dose consequences of design basis accidents, thereby decreasing the calculated thyroid dose consequences of previously evaluated accidents.

Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not modify the configuration of the units, involve any change to plant equipment or change the method of plant operation. The utilization of the ICRP methodology for determining DCFs uses more recent data which only affects calculations for determining thyroid dose consequences.

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

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

The change to utilize the ICRP methodology for determining DCFs allows the use of more recent data which only affects calculations for determining thyroid dose consequences. ICRP–30 is recognized in Revision 1 of NUREG–1432, ''Standard Technical Specifications, Combustion Engineering Plants,'' as an acceptable source document for DCFs. The new methodology will result in more accurate DCFs that will be used in the determination of dose consequences. Utilization of the ICRP–30 DCFs will effectively reduce calculated thyroid dose consequences of design basis accidents, thereby providing additional design margin.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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, Irvine, California 92713.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: September 30, 1998.

Description of amendment request: Revises Units 1 and 2 Technical

Specification (TS) Section 3/4.4.5, ''Steam Generator'' Surveillance Requirements. The installation of the new Delta 94 steam generators at the South Texas Project Units 1 and 2 necessitates changes to the steam generator tube sample selection and inspection requirements; inservice inspection frequencies; acceptance criteria; and inspection reporting requirements.

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

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

Eliminating provisions in the Technical Specifications for applications of the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves for the Delta 94 steam generators is an administrative adjustment, since the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves are not applicable to the Delta 94 steam generators.

The Delta 94 steam generator tubing is designed and evaluated consistent with the margins of safety specified in ASME Code Section III.

The program for periodic inservice inspection of steam generators monitors the integrity of the steam generator tubing to ensure that there is sufficient time to take proper and timely corrective action if tube degradation is present.

The ASME Section XI basis for the 40% through-wall plugging limit is applicable to the Delta 94 steam generators just as it was applicable to the Model E steam generators prior to the implementation of voltage-based repair criteria, F* alternate repair criteria, and laser-welded sleeves. In addition, analysis per Regulatory Guide 1.121 (WCAP–15095/WCAP–15096) has confirmed the applicability of the 40% plugging limit for the Delta 94 steam generators.

The changes also clarify that inservice inspection is required following steam generator replacement, and that inservice inspection is not required during the steam generator replacement outage. This is an administrative change in that it only provides clarification of requirements written without steam generator replacement considerations, and therefore, reduces the possibility for confusion in the application of the subject technical specification provisions. Therefore, these 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.

Eliminating provisions in the Technical Specifications for application of the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves to the Delta 94 steam generators is an administrative adjustment, since the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves are not applicable to the Delta 94 steam generators.

The changes also clarify that inservice inspection is required following steam generator replacement, and that inservice inspection is not required during the steam generator replacement outage. These are administrative changes in that they only provide clarification of requirements written without steam generator replacement considerations, and therefore, reduce the possibility for confusion in the application of the subject technical specification provisions. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Eliminating provisions in the Technical Specifications for applications of the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves for the Delta 94 steam generators is an administrative adjustment, since the voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeves are not applicable to the Delta 94 steam generators.

The Delta 94 steam generator tubing is designed and evaluated consistent with the margins of safety specified in ASME Code Section III. The program for periodic inservice inspection of steam generators monitors the integrity of the steam generator tubing to ensure that there is sufficient time to take proper and timely corrective action if tube degradation is present.

The ASME Section XI basis for the 40% through-wall plugging limit is applicable to the Delta 94 steam generators just as it was applicable to the Model E steam generators prior to the implementation of voltage-based repair criteria, F* alternate repair criteria, and laser-welded sleeves. In addition, analysis per Regulatory Guide 1.121 (WCAP-15095/WCAP–15096) has confirmed the applicability of the 40% plugging limit for the Delta 94 steam generators.

The changes also clarify that inservice inspection is required following steam generator replacement, and that inservice inspection is not required during the steam generator replacement outage. These are administrative changes in that they only provide clarification of requirements written without steam generator replacement considerations, and therefore, reduce the possibility for confusion in the application of the subject technical specification provisions. Therefore, these proposed 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon.

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

Date of application for amendments: September 20, 1996 (TS 96–09).

Brief description of amendments: The amendments would change the Sequoyah Nuclear Plant (SQN) Technical Specifications by clarifying the types of work shifts that are acceptable when considering the requirements to ensure heavy use of overtime is not used routinely by unit staff. The current ''8-hour day'' criteria in Section 6.2.2.g will be expanded to include 10-hour and 12-hour allowances. In addition, the ''40-hour week'' criteria will be changed to a ''nominal 40-hour week'' to provide the necessary flexibility associated with the use of the proposed shift durations.

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

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

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

This change affects the requirements that ensure unit staff personnel do not routinely incur heavy use of overtime. These requirements are not changed by the proposed revision, but are clarified to accommodate the various shift durations used at SQN. The overtime usage by unit staff is not considered to be the initiator for any postulated accident; therefore, the clarification of associated requirements will not increase the probability of an accident. Limiting the use of overtime by staff personnel enhances the operation and maintenance of critical plant equipment that are necessary to mitigate accidents. The proposed revision clarifies these provisions, but does not reduce their adequacy. Therefore, the proposed revision will not increase the consequences of an accident previously evaluated.

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

This change only affects the clarification of shift durations use by unit staff and is not associated with the initiators of accidents. Therefore, the possibility of a new or different kind of accident from any previously analyzed is not created by the proposed clarifications.

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

The proposed changes do not affect plant equipment setpoints or operating policies at SQN. The overtime provisions that ensure the unit staff are capable to operate and maintain the plant in an acceptable manner to provide safe operation and mitigation of accidents is maintained by this change. Therefore, the margin of safety is not reduced by the proposed changes.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, 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: October 2, 1998.

Brief description of amendments: The proposed change would revise Technical Specification (TS) 4.0.6, ''Steam Generator Surveillance Requirements,'' to add definitions required for the F* alternate steam generator tube plugging criterion and identify the portion of the tube subject to the criteria.

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 supporting technical evaluation of the subject criterion [Westinghouse WCAP–15004, listed as Reference 1 (Proprietary)], demonstrates that the presence of the tubesheet enhances the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The result of hardrolling of the tube into the tubesheet is an interference fit between the tube and the tubesheet. A tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings. Analysis and testing have been done to determine the resistive strength of roll expanded tubes within the tubesheet. This evaluation provides the basis for the acceptance criterion for tube degradation subject to the F* criterion. The F* distance of roll expansion is sufficient to preclude tube axial translation or pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The necessary engagement length applicable to the Comanche Peak Unit 1 steam generators is determined to be 1.13 inches, plus an allowance for eddy current measurement uncertainty, based on preload analyses. Verification that this value is significantly conservative was demonstrated by both pullout and hydraulic proof testing. Application of the F* criterion provides a level of protection for tube degradation in the tubesheet region commensurate with that afforded by RG 1.121. Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the F* distance, any postulated faulted condition primary to secondary leakage from F* tubes would be insignificant. No leakage occurred from any of the hydraulic proof test specimens for pressures up to and exceeding faulted condition events. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged.

Based on the above, it is concluded that the proposed F* criterion does not adversely impact any other previously evaluated design basis accidents and operation of Comanche Peak Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed F* criterion does not introduce any significant changes to the plant design basis. Use of the F* criterion does not provide a mechanism to result in an accident initiated outside of the region of the tubesheet expansion. Even if it is postulated that a circumferential separation of a F* tube were to occur below the F* distance, tube structural and leakage integrity will be maintained consistent with the assumptions of the design basis accidents during all plant conditions. Verification of the F* distance of non-degraded tube roll expansion prevents a postulated separated tube from lifting out of the tubesheet during all plant conditions. The F* criterion does not create a possibility for simultaneous failures of multiple tubes. Any other hypothetical accident as a result of any degradation in the expanded portion of the tube would be bounded by the existing steam generator tube rupture accident analysis.

Therefore, it is concluded that the proposed license amendment does 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 use of the F* criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of RG 1.121 (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below the bottom of the transition between the roll expansion and the unexpanded tube or the bottom of the tubesheet (whichever is lower). The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The F* distance has been verified by pullout and hydraulic proof testing of tubes in tubesheet simulating collars to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the FSAR accident analyses.

Implementation of the proposed F* criterion will decrease the number of tubes which must be taken out of service with tube plugs. Plugged tubes reduce the RCS flow margin, thus implementation of the F* alternate plugging criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging.

Therefore, it is concluded that the proposed change does not result in a significant reduction in margin to plant safety as defined in the Final Safety Analysis Report or the bases of the Technical Specifications.

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: John N. Hannon.

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: September 12, 1996, as supplemented April 24, 1997, and September 24, 1998.

Description of amendment request: The staff had previously published a Notice of Consideration of Amendments and Proposed No Significant Hazards Consideration Determination for the licensee's September 12, 1996, application in the **Federal Register** on April 23, 1997 (62 FR 19835). As a result of the staff's requests for additional information, the licensee supplemented its original proposal to relocate the fire protection requirements from the Technical Specifications (TS) to the Updated Final Safety Analysis Report (UFSAR) by letters dated April 24, 1997, and September 24, 1998. The April 24, 1997, letter corrected two minor administrative oversights and does not affect the No Significant Hazards Consideration Determination (NSHCD). However, the September 24, 1998, letter revised the original application to require the Station Nuclear Safety and Operating Committee to submit recommended changes to the offsite review group. In addition, a requirement was added for the establishment, implementation, and maintenance of the Fire Protection Program and implementing procedures. The NSHCD for these changes, as provided in the September 24, 1998, letter, is addressed below.

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:

Since these two changes only deal with administrative requirements, neither of these two specific changes would result in a significant hazards consideration. Therefore, the operation of Surry Power Station with the proposed amendment will not:

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

The probability of an accident is not increased as a result of this Technical Specifications change request. This is an administrative change and merely incorporates two additional requirements for ensuring that the Fire Protection Program and implementing procedures are appropriately established, implemented and maintained, and that changes to the Program and implementing procedures receive the appropriate offsite review. The consequences of an accident previously evaluated are not increased since the station will not be operated differently, and no physical modifications are being made to plant systems or components.

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

A new or different type of accident is not being created since this TS change request is administrative. As noted above, the station will not be operated differently, and no physical modifications are being made to plant systems or components. Administrative revisions regarding the establishment, implementation and maintenance of a TS requirement for a Fire Protection Program and implementing procedures and the imposition of an offsite review for changes thereto [do] not create a new or different type of accident from any accident previously evaluated.

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

The margin of safety as defined in the Technical Specifications is not reduced since system/component performance as assumed in the existing safety analyses is not being affected by the proposed TS change. The TS change is administrative in nature and, as such, has no effect on station operation. The Fire Protection Program is being retained and maintained in the UFSAR and station procedures.

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: 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.

NRC Project Director: Herbert N. Berkow.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Illinois Power Company, Docket, No. 50–461, Clinton Power Station, DeWitt County, Illinois

Date of application for amendment: October 5, 1998.

Brief description of amendment request: The proposed amendment requests deferral of the next scheduled local leak rate test for valve 1MC–042 until the seventh refueling outage.

Date of publication of individual notice in **Federal Register**: October 23, 1998 (63 FR 56949).

Expiration date of individual notice: November 23, 1998.

Local Public Document Room location: Vespasian Warner Public Library, 310 N. Quincy Street, Clinton, IL 61727.

Northeast Nuclear Energy Company, Docket No. 50–423, Millstone Nuclear Power Station, Unit 3, New London County, Connecticut

Date of amendment request: August 6, 1998, as supplemented by letters dated September 3 and 21, 1998.

Description of amendment request: The proposed amendment allows a one-time extension to the steam generator tube inspection surveillance interval until the next refueling outage or July 1, 1999, whichever date is earlier.

Date of publication of individual notice in **Federal Register**: August 17, 1998 (63 FR 43964).

Expiration date of individual notice: September 16, 1998.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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.

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

Date of application for amendment: August 27, 1998, as supplemented by letter dated October 1, 1998.

Brief description of amendment: This amendment revises Technical Specifications (TS) 3.0.4 and 4.0.4 in accordance with the guidance provided in Generic Letter 87–09. The revision to TS 3.0.4 removes the need to explicitly reference its applicability for certain TS. As a result, several other TS were also amended by deleting references to TS 3.0.4.

Date of issuance: October 20, 1998.

Effective date: October 20, 1998.

Amendment No: 84.

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

Date of initial notice in **Federal Register**: September 8, 1998 (63 FR 47529).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1998.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: August 23, 1996.

Brief description of amendments: The amendments revise the Technical Specifications related to the Non-Accessible Area Exhaust Filter Plenum Ventilation System to reflect the design lineup and to make provisions for the performance of maintenance and testing.

Date of issuance: October 15, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 105; 105 & 97; 97.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 12, 1997 (62 FR 11488).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 15, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of amendment request: October 16, 1996, as supplemented by letters dated December 22, 1997, and May 27, 1998.

Brief description of amendment: The amendment changes the Appendix A Technical Specifications by relocating certain administrative controls to Quality Assurance Program Manual as described in Administrative Letter 95–06, ''Relocation of Technical Administrative Controls related to Quality Assurance;'' changing shift coverage from 8-hour day, 40-hour weeks to an option of 8 or 12 hour days and nominal 40-hour weeks; and making editorial changes to the titles of certain organizational positions.

Date of issuance: October 19, 1998.

Effective date: October 19, 1998, to be implemented within 60 days.

Amendment No.: 146.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17233).

The December 22, 1997, and May 27, 1998 letters, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 21, 1995.

Brief description of amendment: The amendments revise the Technical Specification action statements and certain surveillances of TS 3/4.5.1, Safety Injection Tanks (SITs). These revisions include a two-tiered extension of the action completion/allowed outage time for the SITs. The revisions are also consistent with the guidance provided in Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce surveillance requirements for Testing During Power Operation.''

Date of Issuance: October 16, 1998.

Effective Date: To be implemented within 30 days from date of receipt.

Amendment Nos.: 157 and 96.

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

Date of initial notice in **Federal Register**: September 27, 1995 (60 FR 49936).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

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

Date of amendment request: October 31, 1996, supplemented October 31, 1997, May 27, 1998, and September 25, 1998.

Description of amendment request: The amendments revise the administrative control specifications to reduce the administrative burden carried by the Facility Review Group and the Plant General Manager by making more efficient use of site personnel possessing the requisite experience and qualifications in the review and approval process for plant procedures.

Date of Issuance: October 16, 1998.

Effective Date: October 16, 1998.

Amendment Nos.: 158 and 97.

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

Date of Initial Notice in **Federal Register**: December 18, 1996 (61 FR 66707) The October 31, 1997, May 27, 1998, and September 25, 1998, submittals provided clarifying information that did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: August 21, 1998.

Brief description of amendment: The amendment removes the requirement for the Automatic Depressurization System function of the Electromatic Relief Valves to be operable during Reactor Vessel Pressure Testing. Additionally, it clarifies Note h of Technical Specification Table 3.1.1.

Date of Issuance: October 14, 1998.

Effective date: October 14, 1998, to be implemented within 30 days.

Amendment No.: 199.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 10, 1998 (63 FR 48527).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: May 28,1998.

Brief description of amendment: The amendment revises Technical Specification 4.5.A.1 such that the first Type A test required by the primary containment leakage rate testing program be performed during refueling outage 18 rather than refueling outage 17.

Date of Issuance: October 15, 1998.

Effective date: October 15, 1998, to be implemented within 30 days.

Amendment No.: 200.

Facility Operating License No. DPR–16. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38201).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 15, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Illinois Power Company, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois.

Date of application for amendment: May 4, 1998, as supplemented September 23, 1998.

Brief description of amendment: The amendment incorporates Technical Specification requirements for the protection systems for the new static VAR compensators.

Date of issuance: October 9, 1998.

Effective date: October 9, 1998.

Amendment No.: 117.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 3, 1998 (63 FR 30264).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 9, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

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

Date of application for amendment: May 9, 1997, as supplemented August 4, 1998.

Brief description of amendment: The amendment revises the shutdown margin requirements and adds Technical Specification 3/4.3.5 to provide the limiting condition for operation and surveillance requirements for the shutdown margin monitors. The amendment also makes administrative changes and revises the associated Bases section.

Date of issuance: October 21, 1998.

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

Amendment No.: 164.

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

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33129).

The August 4, 1998, letter provided clarifying information that did not change the scope of the May 9, 1997, application, and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 21, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

Date of amendment request: July 11, 1995.

Brief description of amendment: The amendment revises Technical Specifications (TS) 2.3(2)f and 2.3(2)g to increase allowed outage times for the safety injection tanks (SIT).

Date of issuance: October 19, 1998.

Effective date: October 19, 1998.

Amendment No.: 186.

Facility Operating License No. DPR–40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 2, 1995 (60 FR 39447). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 3, 1997, as supplemented by letter dated May 18, 1998.

Brief description of amendment: The amendment revises Technical Specifications (TS) 3.9 to clarify required flow paths for testing the auxiliary feedwater system (AFW) and to delete specific AFW pump discharge pressure.

Date of issuance: October 19, 1998.

Effective date: October 19, 1998, to be implemented 30 days from the date of issuance.

Amendment No.: 187.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63982).

The May 18, 1998, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 1998.

No significant hazards consideration comments received: No.

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

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: July 10, 1998, as supplemented by two letters dated September 11, 1998. The supplemental letters provided clarifying information but did not change the initial no significant hazards consideration determination.

Brief description of amendment: This amendment revises the Technical Specifications for safety limit Minimum Critical Power Ratio from its current value of 1.11 to 1.10 for two recirculation loop operation, and from 1.13 to 1.12 for single recirculation loop operation.

Date of issuance: October 26, 1998.

Effective date: As of date of issuance, to be implemented prior to startup for Cycle 13 operations, scheduled for October 1998.

Amendment No.: 226.

Facility Operating License No. DPR–44: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48261).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 16, 1998, as supplemented by letters dated May 22, August 10, and September 17, 1998, and also by letter dated February 9, 1998.

Brief description of amendments: The amendment authorized changes to the Final Safety Analysis Report to incorporate the increases in the main steam line radiation monitor setpoint and allowable values and the change to the design basis of the offgas system to a detonation resistant design.

Date of issuance: October 13, 1998.

Effective date: October 13, 1998.

Amendment Nos.: 179 and 152.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Final Safety Analysis Report.

Date of initial notice in **Federal Register**: May 20, 1998 (63 FR 27764).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 13, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 23, 1998.

Brief description of amendments: These amendments change the name ‘‘Pennsylvania Power & Light Company’’ to ‘‘PP&L, Inc.’’ in the operating licenses and appendices to reflect the licensee's corporate name change.

Date of issuance: October 19, 1998.

Effective date: Both units, as of the date of issuance to be implemented within 30 days.

Amendment Nos.: 180 and 153.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the operating licenses and Appendix B to each licensee and Attachment 1 to the Unit 1 license.

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35993).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 19, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 25, 1997, as supplemented September 8 and November 18, 1997 and January 8 and July 2, 1998. The supplemental letters provided clarifying information and did not change the initial proposed no significant hazards consideration determination.

Brief description of amendments: These amendments revise the Facility Operating Licenses, Technical Specifications, and Environmental Protection Plans to reflect a corporate name change, remove obsolete information, and correct typographical errors.

Date of issuance: October 23, 1998.

Effective date: Both units, as of date of issuance and shall be implemented within 30 days.

Amendment Nos.: 131 and 92.

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

Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30642).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 23, 1998.

No significant hazards consideration comments received: No.

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

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: June 25, 1997, as supplemented August 3, 1998.

Brief description of amendment: The amendment allows the use of zirconium or stainless steel filler rods in fuel assemblies to replace failed or damaged fuel rods.

Date of issuance: October 8, 1998.

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

Amendment No.: 183.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33107).

The August 3, 1998, submittal fell within the scope of, and did not change, the initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 8, 1997, as supplemented by letters dated March 9, May 6, July 6, July 31, September 4, and September 11, 1998.

Brief description of amendments: The amendments revise the Technical Specifications to accommodate an increase in the maximum licensed thermal power level from 2558 megawatts thermal (MWt) to 2763 MWt.

Date of issuance: October 22, 1998.

Effective date: As of the date of issuance to be implemented on Unit 1 prior to startup from the next refueling outage and on Unit 2 prior to startup from the current refueling outage.

Amendment Nos.: Unit 1–214; Unit 2–155.

Facility Operating License Nos. DPR–57 and NPF–5: The amendments revised the Technical Specifications and Operating Licenses.

Public comments requested as to proposed no significant hazards consideration: Yes. (63 FR 53730 dated October 6, 1998.) The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by November 5, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated October 22, 1998.

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

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

Date of amendments request: May 27, 1997.

Brief Description of amendments: The amendments revise the Technical Specifications (TSs) to change the Applicable Modes for Source Range (SR) Nuclear Instrumentation (NI) (TS ¾.3.1, ''Reactor Trip System Instrumentation''), provide allowances for an exception to the requirements for the state of the power supplies for residual heat removal discharge to charging pump suction valves following Mode changes (TS ¾.5.2, ''ECCS Subsystems—T_{avg}>350°F'' and ¾.5.3, ''ECCS Subsystems—T_{avg}<350°F''), and delete cycle-specific guidance concerning manual engineered safety feature functional input checks.

Date of issuance: October 15, 1998.

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

Amendment Nos.: Unit 1–138; Unit 2–130.

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

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33134).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 15, 1998.

No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: June 5, 1997, as supplemented April 21 and August 12, 1998.

Brief description of amendment: The requested changes would revise the Technical Specifications (TS) to allow testing of diesel generators, pursuant to Surveillance Requirement (SR) 3.8.1.14, during operational modes 1 or 2. The requested changes would also revise the TS to allow testing of the diesel generator batteries and associated battery chargers, pursuant to SRs 3.8.4.12, 3.8.4.13 and 3.8.4.14 during operational modes 1, 2, 3 or 4.

Date of issuance: October 19, 1998.

Effective date: October 19, 1998.

Amendment No.: 12.

Facility Operating License No. NPF–90: Amendment revises the TS.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40561).

The supplemental letter dated August 12, 1998, contained clarifying information and did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 1998.

No significant hazards consideration comments received: None.

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

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

Date of application for amendment: April 8, 1998, as revised by letter dated August 27, 1998.

Brief description of amendment: The amendment reduces the allowable reactor coolant system specific activity from 1.0 microcurie/gram to 0.20 microcurie/gram dose equivalent I–131, a means described by Generic Letter 95–05 to support the reduction of reactor coolant system specific activity limits.

Date of issuance: October 27, 1998.

Effective date: October 27, 1998.

Amendment No.: 140.

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

Date of initial notice in **Federal Register**: September 14, 1998 (63 FR 49137).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1998.

No significant hazards consideration comments received: No.

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

Notice of Issuance of Amendment to Facility Operating License and Final No Significant Hazards Consideration Determination

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

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 December 4, 1998, 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 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of 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 50–530, Palo Verde Nuclear Generating Station, Unit No. 3, Maricopa County, Arizona

Date of application for amendment: October 6, 1998

Brief description of amendment: The amendment revises TS 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operation,'' and TS 3.3.2, ''Reactor Protective System (RPS) Instrumentation—Shutdown.'' The proposed amendment would clarify the power level threshold at which certain RPS instrumentation trips must be enabled and may be bypassed, and would clarify that this level is a percentage of the neutron flux at rated thermal power (RTP). The bypass power level, 1E–4% RTP, would be specified as logarithmic power instead of thermal power.

Date of issuance: October 19, 1998.

Effective date: October 19, 1998.

Amendment No.: 119.

Facility Operating License No. NPF–74: The amendment revised the Technical Specifications.

Press release issued requesting comments as to proposed no significant hazards consideration: Yes. October 13, 1998. Arizona Republic Newspaper (Arizona).

Comments received: No. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Arizona and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 19, 1998.

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.

Dated at Rockville, Maryland, this 28th day of October 1998.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[FR Doc. 98–29433 Filed 11–3–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 23511; 812–11252]

FPA Capital Fund, Inc.; Notice of Application

October 29, 1998.

AGENCY: Notice of application under section 17(b) of the Investment Company Act of 1940 (''Act'') for an exemption from section 17(a) of the Act.

SUMMARY OF APPLICATION: Applicant, FPA Capital Fund, Inc. (''Fund''), seeks an order to permit an in-kind redemption of shares of the Fund by an affiliated person of the Fund.

FILING DATES: The application was filed on August 6, 1998 and amended on October 20, 1998.

Hearing or Notification of Hearing: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on November 23, 1998, and should be accompanied by proof of service on applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the Commission's Secretary.

ADDRESSES: Secretary, Commission, 450 Fifth Street, NW., Washington, DC 20549. Applicants, 11400 West Olympic Boulevard, Los Angeles, California 90064.

FOR FURTHER INFORMATION CONTACT: Deepak T. Pai, Attorney-Adviser, (202) 942–0574 or Edward P. Macdonald, Branch Chief, at (202) 942–0564 (Division of Investment Management,