oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Dr. John T. Larkins (telephone: 301/415–7360) 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 changes in schedule, etc., that may have occurred.

Dated: January 8, 1998.

Gail H. Marcus,

Acting Deputy Executive Director.

[FR Doc. 98–873 Filed 1–13–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Advanced Reactor Designs; Notice of Meeting

The ACRS Subcommittee on Advanced Reactor Designs will hold a meeting on February 3 and 4, 1998, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

Portions of the meeting may be closed to public attendance to discuss Westinghouse Electric Corporation proprietary information related to the Test and Analysis Program pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Tuesday, February 3, 1998—1:00 p.m. until the conclusion of business
Wednesday, February 4, 1998—8:30 a.m. until the conclusion of business

The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and Westinghouse regarding the AP600 Test and Analysis Program and the AP600 Standard Safety Analysis Report Chapters 1, 4, 5, 7, 8, 9, 10, 11, 13, 17 and 18. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

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

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

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

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Noel F. Dudley (telephone 301/415–6888) 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: January 8, 1998.

Gail H. Marcus,

Acting, Deputy Executive Director.

[FR Doc. 98–948 Filed 1–13–98; 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 from December 18, 1997, through January 2, 1998. The last biweekly notice was published on December 31, 1997 (62 FR 68303).

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

By February 13, 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, STN 50–529, and STN 50-530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of amendments request: December 17, 1997.

Description of amendments request: The proposed amendment would modify the Technical Specifications (TS) to replace the current explicit reference to Exide batteries with a generic reference to low specific gravity cells. The proposed change would also remove footnotes for the Unit 2 and Unit 3 TS that referred to one time exemptions that no longer apply. The proposed change would allow replacement of the existing Class 1E, 125 volt DC batteries with equivalent batteries manufactured by different vendors.

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 Class 1E 125V DC system provides DC power to the Class 1E DC loads for operation, control and switching, including the inverters which power the Class 1E 120V vital AC busses. This system is not an accident initiator. It is, however, an accident mitigation system. The replacement low specific gravity rectangular cell batteries have been designed to IEEE 485–1978 standards and meet all appropriate seismic criteria. There is no change in the physical or electrical separation provisions for the Class 1E 125V DC channels. These batteries are used extensively throughout the industry and their failure mechanisms are well understood. The existing high specific gravity round cell batteries are experiencing premature capacity loss for which a definitive root cause of failure has not been determined. Therefore, replacement of the high specific gravity round cell batteries with low specific gravity rectangular cell batteries increases the overall reliability of the Class 1E 125 V DC system. In addition, the design requirements of the replacement batteries ensures that the batteries will be capable of reliably performing their design function during all modes of operation and will serve to mitigate any accident that may occur. The proposed amendment does not change the performance criteria or cell parameters for the Class 1E 125V DC sources that are defined in the current Technical Specifications for each unit. Since this change is increasing the overall reliability and performance of the system and is designed to meet the same stringent requirements of the existing high specific gravity round cell batteries, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Replacement of the high specific gravity round cell batteries will occur during acceptable modes of operation as defined in the current Technical Specifications for each unit, i.e., the work will be performed during Modes 5 or 6, or with the reactor defueled. Technical Specification 3.8.2.2, DC Sources—Shutdown, for each unit requires one Class 1E 125V DC train to be operable in Modes 5 or 6. With one Class 1E 125V DC train operable, the other train may be removed from service for battery cell replacement. Since the battery cell replacement will be performed within the Limiting Condition of Operation for DC Sources—Shutdown, the replacement sequence of the battery banks will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Low specific gravity rectangular cell batteries have been used throughout the industry for many years. Some cells have been in service for 17 years and have not degraded to the extent that they require replacement. The low specific gravity rectangular cell batteries have demonstrated good reliability and the failure mechanisms associated with these batteries are well understood. The high specific gravity round cell batteries that are currently installed are exhibiting premature capacity loss for which a definitive root cause of failure has not been determined. Replacing the high specific gravity round cell batteries with low specific gravity rectangular cell batteries that have seen extensive use in the industry, are well understood and have been designed to meet the same stringent requirements as that of the existing batteries ensures that the overall system reliability is increased. No new or common mode failures are created since the replacement low specific gravity rectangular cell batteries have been designed to the same stringent requirements as the existing batteries. The proposed amendment does not change the performance criteria or cell parameters for the Class 1E 125V DC sources that are defined in the current Technical Specifications for each unit. Therefore, replacement of the high specific gravity round cell batteries with low specific gravity rectangular cell batteries 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.

As described previously, replacing the high specific gravity round cell batteries with low specific gravity rectangular cell batteries enhances the overall system reliability. The low specific gravity rectangular cell batteries have been designed to the same criteria as the existing high specific gravity round cell batteries. The performance criteria and cell parameters specified in each unit's Technical Specifications are not affected by this change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Replacement of the high specific gravity round cell batteries will occur during acceptable modes of operation as defined in the current Technical Specifications for each unit, i.e., the work will be performed during Modes 5 or 6, or with the reactor defueled. Technical Specification 3.8.2.2, DC Sources—Shutdown, for each unit requires one Class 1E 125V DC train to be operable in Modes 5 or 6. With one Class 1E 125V DC train operable, the other train may be removed from service for battery cell replacement. Since the battery cell replacement will be performed within the Limiting Condition of Operation for DC Sources—Shutdown, the work sequence for replacement of the battery banks will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involve 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.

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

Date of amendment request: October 3, 1996.

Description of amendment request: The proposed amendments would correct a typographical error which was introduced into the Technical Specifications (TS) with issuance of Amendment Nos. 150 and 145.

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

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

The proposed change does not alter the manner of operation of the facility, it merely restores the correspondence between the applicability of the Limiting Conditions for Operability (LCO) for the (Source Range) Neutron Monitors and the Drywell Radiation Monitors and the associated Surveillance Requirements for the same two instrument functions as described in Tables 3.2.F–1 and 4.2.F–1.

No changes are proposed which will affect the probability of an accident previously evaluated, since the instruments and their associated functions are credited to operate during and after a postulated accident. The function of a device after an event has occurred cannot affect the probability of that accident occurring. Similarly, the proposed changes do not effect the operation or function of structures, systems or components which effect the probability of any accident previously evaluated.

The proposed changes do not affect the consequence of an accident previously evaluated since the changes do not decrease the availability of any functions credited with performing mitigative actions. The availability requirements of the Drywell Radiation Monitors is not changed because the associated LCO requires the monitors to be OPERABLE in the conditions proposed in this change. The proposed change merely assures that the surveillance requirements are met in the modes which correspond to the LCO. The (Source Range) Neutron Monitor surveillance requirements change does not affect the ability of the system to provide adequate information to the operators to mitigate the consequences of a postulated accident, since the system OPERABILITY requirements as specified in the LCO are not affected.

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

The proposed change does not introduce any new or different types of operation of the plants. No new equipment is introduced as a result of the implementation of the proposed change. Therefore no changes are proposed which could introduce a new or different kind of accident from any previously evaluated.

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

The proposed change does not effect the margin of safety. The LCO requirements for the two instrument systems which are effected are not changed; the OPERABILITY requirements remain the same. The only substantive changes are the modes in which surveillance testing is required to be performed. The change restores the need to perform testing of the Drywell Radiation Monitor prior to and during OPERATIONAL MODE 3 operations, and removes the requirement to perform testing of the (Source Range) Neutron Monitors prior to and during operation in MODE 3 when it is not required to be OPERABLE as described in the associated LCO. Based on this, the availability of the affected instruments to perform their design function is not effected by this change and no reduction in the margin of safety is proposed.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

This proposed amendment does not involve any irreversible changes, significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the **Federal Register** and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

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

NRC Project Director: Robert A. Capra.

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: August 29, 1997.

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 Siemens' methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates, (b) administrative changes that both eliminate the cycle specific implementation of Atrium-9B fuel, and (c) changes to the Dresden and Quad Cities Minimum Critical Power Ratio (MCPR).

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.

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

The Reference 1 [ANF–91–048(P), Supplement 1, ''BWR Jet Pump Model Revision for RELAX''. Submitted to the NRC by SPC letter, ANF–91–048(P), Supplement 1 and ANF–91–048(NP), Supplement 1, ''BWR Jet Pump Model Revision for RELAX,'' RAC:96–042, R.A. Copeland to US NRC, May 6, 1996] 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.

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 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] 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.

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 is not increased by adding Reference 7 [ANF–1125(P), Supplement 1, Appendix D, ''ANFB Critical Power Correlation Uncertainty For Limited Data Sets''. Submitted to the NRC by SPC letter, ''Request for Review of ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF–1125(P), Supplement 1, Appendix D'', HDC:97:032, H.D. Curet to Document Control Desk, April 18, 1997] 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 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 more than 99.9% 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 of consequences of an accident previously evaluated are not affected by these changes.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit at Quad Cities Units 1 and 2 and Dresden Units 2 and 3 will not increase the probability of an accident previously evaluated. This change implements the MCPR Safety Limits resulting from the SPC ANFB critical power correlation methodology using a revised additive constant uncertainty from Reference 7. The MCPR Safety Limit of 1.09 that is proposed for Quad Cities Units 1 and 2 and Dresden Units 2 and 3 is 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. Changing the MCPR Safety Limit will not alter any physical systems or operating procedures. The MCPR Safety Limit is set to 1.09, which is the CPR value where less than 0.1% of the rods in the core are expected to experience boiling transition. This safety limit is expected to be applicable for future cycles of ATRIUM–9B at Dresden and Quad Cities. Therefore the probability or consequences of an accident will not increase.

Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 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 is not expected to increase.

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 average planar linear heat generation rate] limits and their exposure basis to be specified in the COLR [core operating limits report] and to establish consistency with approved methodologies currently utilized by Siemens Power Corporation, which calculates 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 [boiling water reactor]. 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 wording. The definition of the Average Planar Exposure is deleted, because the exposure basis of the APLHGR 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.

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.

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.

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. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

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% of the rods avoid boiling transition.

The MCPR Safety Limit is changing for Quad Cities Unit 1 due to the transition to SPC ATRIUM–9B fuel and SPC methodologies. The MCPR Safety Limit is changing for Quad Cities Unit 2 due to the Reference 7 methodology, which documents a 0.0195 ATRIUM–9B additive constant uncertainty and supports a 1.09 MCPR Safety Limit. This MCPR Safety Limit is lower than the current MCPR Safety Limit for Quad Cities Unit 2, 1.10, which is based on a higher interim conservative additive constant uncertainty of 0.029. The lower ATRIUM–9B additive constant uncertainty results in the lower MCPR Safety Limit for Quad Cities Unit 2. The new MCPR Safety Limit for Dresden Units 2 and 3, 1.09, is greater than the current value at Dresden Units 2 and 3 and is being increased now in anticipation of bounding future reloads of ATRIUM–9B. Therefore, no new accidents are created that are different from any accident previously evaluated.

Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 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 the generic Reference 3 methodology and the anticipated approval of the Reference 7 additive constant uncertainty methodology, these footnotes are no longer applicable. 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.

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.

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

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.

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 this reference 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. 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 topical 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–95–021(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] that greater than 99.9% 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.

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% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. This Technical Specification amendment 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 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 only ensure that the appropriate level of fuel protection is being employed.

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

Changing the MCPR Safety Limit for Quad Cities and Dresden 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% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. The proposed Technical Specification amendment reflects the MCPR Safety Limit results from conservative evaluations by SPC using the ANFB critical power correlation with the new 0.0195 ATRIUM–9B additive constant uncertainty documented 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% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 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) have been submitted to the NRC and both are proposed to be added to the Technical Specifications in this amendment, 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.

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 La Salle, Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Project Director: Robert A. Capra.

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

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

Date of amendment request: October 27, 1997.

Description of amendment request: The proposed amendments would clarify the applicability, action and surveillance requirements for the Standby Liquid Control 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.

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

The proposed changes represent the conversion of current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed changes are consistent with NUREG–1433 and do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and represents sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. The proposed TS continue to ensure sufficient requirements are in place for the SLCS during plant operation. The proposed changes that eliminate Applicability and Actions during refueling operations for the SLCS do not affect the probability of any previously evaluated accident because only one control rod can be withdrawn during refueling operations and Shutdown Margin requirements are maintained in the Technical Specifications. Therefore, the probability of an inadvertent criticality is not increased as reactivity controls are maintained. Because the SLCS is manually initiated and not assumed to mitigate any accident scenario during refueling operations, the proposed changes do not affect the consequences of any previously evaluated accident. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the revisions proposed to the surveillance requirements are administrative in nature and either relocate procedural details to administrative controls or allow provisions for manual alignment of a manual system to the proper orientation. As such, because there is no effect on any accident scenario, the probability of any accident previously evaluated is not increased by the proposed amendment. Because the proposed changes are administrative in nature, the consequences of any previously evaluated accident are not increased.

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

The proposed amendment for Dresden and Quad Cities Station's Technical Specification is based on generic guidance or NRC accepted changes for later operating BWR plants. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the generic guidance. The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. SLCS requirements are adequately retained to ensure sufficient controls remain during plant operations. The proposed changes to the Applicability and Actions during refueling operations for the SLCS do not create a new or different kind of previously evaluated accident. Because the SLCS is manually initiated to mitigate accident concerns during power operations, the proposed deletion of Applicability and Actions during refueling operations does not affect the probability of a new or different kind of accident from being created. The changes proposed to the surveillance requirements are administrative in nature and do not affect the system operation; as such, the proposed changes do not affect the probability of a new or different kind of accident being created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment represents the conversion of current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed changes are consistent with NUREG–1433 and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Dresden or Quad Cities based on system design, safety analysis requirements and operational performance. SLCS provisions continue to be adequately maintained during plant operation. The proposed changes to the Applicability and Actions during refueling operations for the SLCS do not significantly reduce existing plant safety margins. Because the SLCS is manually initiated to mitigate accident concerns during power operations, the proposed deletion of Applicability and Actions during refueling operations has no effect on existing plant safety margins as this system is not required during this mode of operation. The changes proposed to the surveillance requirements are administrative in nature and do not affect the system operation; as such, the proposed changes do not adversely affect existing plant safety margins as adequate system surveillance requirements are maintained. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden or Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute 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 requested amendments involve 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.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: November 7, 1997.

Description of amendment request: The proposed amendments would relocate the Unit 2 24/48 Vdc batteries, chargers, and distribution systems operability and surveillances requirements from the Technical Specifications to licensee administratively controlled documents.

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

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

Removal of the Unit 2 24/48 Vdc battery, charger, and distribution panel requirements from the Technical Specification requirements of 3/4.9.C, 3/4.9.D, 3/4.9.E, and 3/4.9.F and the subsequent relocation of those requirements to licensee administrative controls is an administrative change that will continue [to] ensure the availability of the Unit 2 24/48 Vdc system and will not increase the probability of accidents previously evaluated. Relocation of the Unit 2 24/48 Vdc requirements to administrative controls will have no effect on the control instrumentation and cannot act as an initiator for any of the accidents evaluated in the UFSAR [Updated Final Safety Analysis Report].

Similarly, relocation of the Unit 2 24/48 Vdc system requirements to licensee administrative controls will have no effect on the availability of the loads which are supplied by the Unit 2 24/48 Vdc batteries nor on any of the consequences of accidents previously evaluated in the UFSAR. Control of the Unit 2 24/48 Vdc requirements by licensee administrative controls under 10 CFR 50.59 will not affect any of the protection or mitigation functions which may be provided by any of the loads supplied by the batteries. Operation under the proposed amendment will not significantly increase the probability or consequences of any accidents previously evaluated.

Because of the above evaluation, removal of the Unit 2 24/48 Vdc system from the Technical Specifications will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The Unit 2 24/48 Vdc batteries, chargers, and other components will retain the separation, and redundancy under which they are presently installed. No new failure modes are introduced by this administrative relocation of requirements, for the Unit 2 24/48 Vdc system, from the Technical Specifications to licensee administrative control. Since the batteries are not being operated differently and transferring ATS [Analog Trip System] loads to the 125 Vdc safety-related battery system does not affect the function or mode of operation of these loads, the possibility of a new or different accident from any accident previously evaluated is not increased or created by this administrative change.

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

Relocation of the TS requirements for the Unit 2 24/48 Vdc system does not affect the operating points or setpoints of any systems or components. Plant operating points or parameters are not changed by the proposed relocation of requirements in this amendment request. The safety-related equipment that is supported by the Unit 2 24/48 Vdc system will continue to be required in the existing modes of applicability as determined by the individual equipment Technical Specifications. Thus operation under the proposed license amendment removes some redundancy and constraints during refueling but does 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

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

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket No. 50–373, LaSalle County Station, Unit 1, LaSalle County, Illinois

Date of amendment request: November 24, 1997.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.3.2, ''Isolation Actuation Instrumentation,'' to add instrumentation for the reactor water cleanup (RWCU) pump rooms and valve room as a result of modifications to the RWCU system. Also, additional instrumentation will be added in the RWCU holdup pipe area, the filter/demineralizer valve rooms, and RWCU pump suction high flow switch as a result of a high energy line break re-evaluation. The setpoints for the RWCU heat exchanger room instrumentation will be revised as a result of new design basis calculations. The proposed amendment will also delete instrumentation related to the residual heat removal (RHR) steam condensing mode which is no longer utilized and will eliminate the alarm and isolation functions for the RHR shutdown cooling mode.

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

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

(a) There is no effect on accident initiators so there is no change in [the] probability of an accident. A line break in the subject areas would consist of an instantaneous circumferential break downstream of the outermost isolation valve of one of these systems. The leak detection isolation is only a precursor of a break, and thus does not affect the probability of a break.

(b) There is no or minimal effect on the consequences of analyzed accidents, due to changing the leak detection ambient T or Delta T setpoint and allowable values to detect 25 gpm equivalent leakage. The addition of more ambient T and Delta T leak detection monitoring, along with the addition of the high flow break detection will actually decrease the consequences of the associated accidents. The worst case accident outside the primary containment boundary is a main steam line break which bounds the dose consequences of all line breaks and therefore bounds any size of leak.

The deletion of the RHR steam condensing mode isolation actuation instrumentation trip functions from the LaSalle TS does not increase the probability or consequences of an accident previously evaluated, because this mode of operation of the RHR system has been deleted from the LaSalle design basis and the lines that were previously high energy line are isolated during unit operation, including Operational Condition 1 (Run mode), Operational Condition 2 (Startup mode), and Operational Condition [3] (Hot Shutdown).

The deletion of the RHR shutdown cooling mode leak detection T and Delta T isolation actuation instrumentation trip functions from the LaSalle TS does not increase the probability or consequences of an accident previously evaluated, because the leak detection is only a precursor of a break, and thus does not affect the probability of a break. Also, there are two remaining different methods of detecting abnormal leakage and isolating the system in technical specification trip functions A.6.a, Reactor Vessel Water Level—Low, Level 3 and A.6.c, RHR Pump Suction Flow—High. In addition, other means to detect leakage from the RHR system, such as sump monitoring and area radiation monitoring, are also available. In accordance with TS Administrative Requirement 6.2.F.1, LaSalle has a leakage reduction program to reduce leakage from those portions of systems outside primary containment that contain radioactive fluids. RHR, including piping and components associated with the shutdown cooling mode, is part of this program, which includes periodic visual inspection for system leakage. The sump monitoring, radiation monitoring and periodic inspections for system leakage makes the probability of a leak of 5 gpm going undetected for more than a day very low.

Also, due to the low reactor pressures (less than 135 psig) at which RHR shutdown cooling mode is able to operate, reactor coolant makeup and outflow is very low compared to normal plant operation. A change in flow balance due to a leak is thus more readily detectable with reactor coolant water level changes and makeup flow rate, and thus precludes a significant leak going undetected before break detection instrumentation would cause automatic isolation.

Therefore, there is not a significant increase in the probability or consequences of an accident previously evaluated.

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

The purpose of the leak detection system, as it applies to the RWCU and RHR system areas, is to provide the capability for leak detection and automatic isolation as necessary of the system in the event of leakage in these areas. This change maintains this capability with at least two different methods of detection of abnormal leakage for protection from the flooding concerns of a significant leak or line break when the RHR system is operating in the shutdown cooling mode, so that redundant systems will not be affected.

This change also maintains or adds primary containment isolation logic for the leak detection isolation based on temperature monitoring in RWCU areas and break detection based on RWCU pump suction flow—high. The additional instrumentation and the associated isolation logic is the same or similar to existing instrumentation and logic for containment isolation actuation instrumentation, so no new failure modes are created in this way.

Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

The change to the automatic isolation setpoint for high Delta T leak detection in the heat exchanger rooms is based on current configuration calculated/analyzed response to a small leak compared to a circumferential break. The increased leakage rate in the RWCU heat exchanger rooms that is necessary to actuate isolation on high temperature during winter conditions, does not adversely affect the margin of safety. This increased leakage rate is below the critical crack leakage rate as represented in [Updated Final Safety Analysis Report] UFSAR Figure 5.2–11. Additionally, differential temperature leak detection is conservative under these same conditions, and will actuate isolation at a leakage rate less than the established limit. The leak detection isolation logic is unchanged and thus remains single failure proof.

The addition of automatic primary containment isolation on Ambient and Differential Temperature (Delta T)-High for the Reactor Water Cleanup System (RWCU) Pump, Pump Valve, Holdup Pipe, and Filter/Demineralizer (F/D) Valve Rooms and the addition of the RWCU Pump Suction Flow High line break isolation add to the margin of safety with respect to leak detection and line breaks in the RWCU system, because the system isolation diversity is increased and the amount of system piping monitored for leakage is increased.

The setpoints for the ambient temperature and differential temperature leak detection isolations being changed or added and the RWCU pump suction flow—high are set sufficiently high enough so as not to increase the possibility of spurious actuation. In the event that a spurious actuation does occur, little safety significance is presented since the RWCU system performs no safety function. The setpoints and allowable values for the proposed changes also assure sufficient margin to the analytical values and [are] high enough to prevent spurious actuations based on calculations consistent with Regulatory Guide 1.105.

The deletion of the RHR steam condensing mode isolation actuation instrumentation does not effect the margin of safety, because this mode is no longer utilized by LaSalle in Operational Conditions 1, 2, or 3 (Run mode, Startup Mode, or Hot Shutdown).

The elimination of the temperature based trip functions for the RHR shutdown cooling mode area is based on the determination that temperature is not the appropriate parameter as it does not provide meaningful indication and will not provide setpoints that would be sufficiently above the normal range of ambient conditions to avoid spurious isolations.

There are two remaining different methods of detecting abnormal leakage and isolating the system in technical specification trip function A.6, namely A.6.a, Reactor Vessel Water Level—Low, Level 3 and A.6.c, RHR Pump Suction Flow—High. In addition, other means to detect leakage from the RHR system, such as sump monitoring and area radiation monitoring, are also available. Also, in accordance with TS Administrative Requirement 6.2.F.1, LaSalle has a leakage reduction program to reduce leakage from those portions of systems outside primary containment that contain radioactive fluids. RHR, including piping and components associated with the shutdown cooling mode, is part of this program, which includes periodic visual inspection of system for leakage.

The previous evaluation of diversity of isolation parameters, as presented in Table 5.2–8 of the UFSAR remains unchanged. Adequate diversity of isolation parameters is maintained because there are at least two different methods available to detect and allow isolation of the system for a line break, as necessary.

Therefore, this requested Technical Specification amendment does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: December 10, 1997 (NRC–97–0105).

Description of amendment request: The proposed amendment would modify the technical specifications (TS) and the bases to accommodate the installation of an improved power range neutron monitoring system. The modification and the TS changes represent part of the licensee's actions in response to Generic Letter 94–02, ''Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors,'' dated July 11, 1994. The TS revisions include changes to Action Statements and Surveillance Requirements which are generally consistent with licensing topical report NEDC–32410P–A, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function,'' and NEDC–32410P Supplement 1, ''NUMAC PRNM Retrofit Plus Option III Stability Trip Function,'' which were reviewed by the NRC as documented in a letter dated September 5, 1995, and a safety evaluation dated August 15, 1997. The proposed amendment also includes two unrelated changes. Surveillance Requirement 4.3.1.3 and its associated bases are modified to clarify the applicability of response time testing requirements. In addition, the first page of Table 3.3.6–2 is modified to correct a typographical error in the title.

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

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

This proposed TS change is associated with the NUMAC–PRNM retrofit design. The proposed TS change involves modification of the Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) for equipment designed to mitigate events that result in power increase transients. The APRM [average power range monitor] system mitigative action is to block control rod withdrawal or initiate a reactor scram, which terminates the power increase when setpoints are exceeded. The Rod Block

Monitor (RBM) system mitigative action is to block continuous control rod withdrawal prior to exceeding the fuel design limits during a postulated Rod Withdrawal Error. The functional capability of the previous Reactor Coolant System Recirculation Flow control rod block trip functions have been incorporated into the modified APRM control rod block trip functions. The worst case failure of either the APRM or the RBM systems is failure to initiate mitigative action (failure to scram or block rod withdrawal). Failure to initiate mitigative action will not increase the probability of an accident. Thus, the proposed change does not increase the probability of an accident previously evaluated.

For the APRM and the RBM systems, the NUMAC PRNM design, together with revised operability requirements (LCOs) and revised testing requirements (SRs), continues to perform the same mitigation functions under identical conditions with availability comparable to the types of equipment that it replaces. Because there is no change in mitigation functions and because availability of the functions is maintained, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes involve modification and replacement of the existing power range neutron monitoring equipment, and modification of the setpoints and operational requirements for the APRM and RBM systems. These proposed changes do not modify the basic functional requirements of the affected equipment, create any new system interfaces or interactions, nor create any new system failure modes or sequence of events that could lead to an accident. The worst case failure of the affected equipment is failure to perform a mitigation action, and failure of this mitigative equipment does not create the possibility of a new or different kind of accident. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS change is associated with the NUMAC PRNM retrofit design. The NUMAC PRNM change does not impact reactor operating parameters nor the functional requirements of the power range neutron monitoring system. The replacement equipment continues to provide information, enforce control rod blocks and initiate reactor scrams under appropriate specified conditions. The proposed change does not revise any safety margin requirements. The replacement APRM/RBM equipment has improved channel trip accuracy compared to the current system and meets or exceeds system requirements previously assumed in setpoint analysis. Thus, the ability of the new equipment to enforce compliance with margins of safety equals or exceeds the ability of the equipment which it replaces. The proposed change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. The editorial change in Table 3.3.6–2 and the clarification in Surveillance Requirement 4.3.1.3 also satisfy the three standards of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

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: December 1, 1997.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to add a time delay, including allowance, to a portion of the Engineered Safety Feature Actuation System undervoltage (UV) trip TSs. The proposed changes would result in the TSs being consistent with the current design, as detailed in the Final Safety Analysis Report, and the current surveillance procedures.

Specifically, TS Table 3.3–4, Loss of Power, would be changed by adding a 2.0 [plus or minus] 0.1 second time delay for the 4.16 kV Emergency Bus UV (UV Relays) level 1—Trip Setpoint and the Allowable values.

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

NNECO concludes that these proposed additions to Technical Specification Table 3.3–4 do not involve a significant hazards consideration (SHC) and do not involve a significant impact on public health and safety. The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. That is, the proposed changes do not:

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

The proposed changes will add a time delay, including allowance, to a portion of the Engineered Safety Feature Actuation System (ESFAS) Undervoltage (UV) Trip Technical Specification Table 3.3–4. These changes will align the Technical Specifications to the existing plant design, as described in the Final Safety Analysis Report (FSAR) system description and the existing surveillance procedure. No new plant modifications are associated with this addition to the Technical Specifications.

The addition of the Level One UV trip time delay setpoint does not impact any system or component whose failure results in initiation of the accidents described in the FSAR. Therefore, the changes do not affect the probability of occurrence of the previously evaluated accidents. The Level One UV trip time delay potentially affects the Emergency Diesel Generator (EDG) response time to accident conditions that occur coincident with a loss of normal power (LPN). However, previous analysis of the increase in the time delay (0.5 seconds to 2.0 [plus or minus] 0.1 second) concluded that the ESFAS response times for those events considered to occur coincident with an LNP, are not challenged by the time delay. This conclusion is based upon a comparison between the EDG start time and the maximum time required to complete those LNP trip functions necessary to support EDG availability for worst case accident conditions (Loss of Coolant Accident which results in a Safety Injection Actuation Signal (SIAS) coincident with LNP). The calculated EDG start time considered the ESFAS response time (0.5 seconds) in addition to the maximum EDG start time of 15 seconds after receipt of an SIAS, as specified in Technical Specification Surveillance Requirement 4.8.1.1.2.a.2. Since the calculated LNP trip time delay of 15.14 seconds is less than the calculated SIAS initiated EDG start time of 15.5 seconds, the proposed changes do not increase the likelihood of an EDG malfunction during an accident condition. Consequently, the proposed additions do not adversely affect the ability of either the ESFAS or the EDGs to perform their intended safety function. The proposed additions to Table 3.3–4 do not modify the Limiting Condition for Operation or the specific surveillance procedure acceptance criterion, nor do they change the frequency of the surveillance. The proposed changes do not involve any physical changes to the plant and do not alter the way any structure, system, or component functions. The proposed changes do not have any adverse impact on the design basis accidents previously analyzed. The proposed changes do not result in an increase in radiation exposure to either members of the public or site personnel because accident mitigation systems will be available consistent with the assumptions used in the accident analysis. Therefore, the proposed additions to Technical Specification Table 3.3–4 do not affect the consequences of the previously evaluated accidents.

Based on the above, the proposed changes do not involve a significant increase in the probability of consequences of an accident previously evaluated.

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

The function availability and failure modes of equipment important to safety are unaffected by the addition of the 2.0 [plus or minus] 0.1 second Level One UV trip time delay to Technical Specification Table 3.3–4. The additions do not introduce any new, credible accidents, or any new failure modes.

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

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

The proposed additions to Technical Specification Table 3.3–4 do not have any adverse impact on the accident analyses. Actuation of the required safety systems is not delayed because the proposed additions do not delay the time at which the EDGs are required, by the plant Technical Specifications, to be available to power the required loads.

Therefore, based on the above, there is no significant reduction in the margin of safety.

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

Local Public Document Room location: 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 Deputy Director: Phillip F. McKee.

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

Date of amendment request: December 15, 1997.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) to adopt Option B, of 10 CFR Part 50, Appendix J, to implement a performance-based approach for Type B and C testing. Additionally, the wording in the TSs would be modified for the previous adoption of Option B on Type A testing and a section added on the primary containment leakage rate testing program.

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

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

Containment leak rate testing is not an initiator of any accident. The proposed changes do not make any physical changes to the containment and do not affect reactor operations or the accident analyses. Therefore, the proposed changes do not involve a significant increase in the probability of any previously evaluated accident.

Since the allowable leakage rate is not being changed and since the analysis documented in NUREG–1493, ''Performance-Based Containment Leak-Rate Program'' concludes that the impact on public health and safety due to extended intervals is negligible, the proposed changes will not involve a significant increase in the consequences of any previously evaluated accident.

Therefore, adoption of a performance-based leakage testing requirements will provide an equivalent level of safety and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No physical changes are being made to the plant, nor are there any changes being made to the operation of the plant as a result of the proposed changes. In addition, no new failure modes of plant equipment previously evaluated are being introduced.

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

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

The proposed changes are based on NRC-approved provisions and maintain adequate levels of reliability of containment integrity. The performance-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity. This supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible. Since the analysis documented in NUREG–1493 confirms that the performance based schedule continues to maintain a minimal impact on public risk, it can be concluded that the margin of safety is not significantly affected by the proposed changes.

Therefore, the proposed amendment 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: 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: John F. Stolz.

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

Date of amendments request: July 23, 1997, as supplemented September 30 and December 18, 1997. The July 23, 1997, application was previously noticed in the **Federal Register** on September 10, 1997 (62 FR 47699). The December 18, 1997, supplement provided additional information that revised the licensee's evaluation of the significant hazards consideration. Therefore, renotification of the Commission's proposed determination of no significant hazards consideration is necessary.

Description of amendments request: The proposed amendments would revise the Technical Specifications (TSs) by relocating the reactor coolant system pressure and temperature limits from the TSs to the proposed Pressure Temperature Limits Report in accordance with the guidance provided by Generic Letter 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.''

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

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

The proposed removal of the Reactor Coolant System (RCS) pressure temperature (P–T) limits from the Technical Specifications (TSs) and relocation to the proposed Pressure Temperature Limits Report (PTLR) in accordance with the guidance provided by Generic Letter (GL) 96–03 is administrative in that the requirements for the P–T limits are unchanged. The P–T limits proposed for inclusion in the PTLR are based on the fluence associated with 2775 MW thermal power and operation through 21.9 effective full power years (EFPY) for Unit 1 and 33.8 EFPY for Unit 2. GL 96–03 requires that the P–T limits be generated in accordance with the requirements of 10 CFR [Part] 50, Appendices G and H, documented in an NRC-approved methodology incorporated by reference in the TSs.

Accordingly, the proposed curves have been generated using the NRC-approved methods described in WCAP–14040–NP–A, Revision 2, as modified at the direction of the NRC Staff, and meet the requirements of 10 CFR [Part] 50, Appendices G and H. TS 3.4.10.1 will continue to require that the RCS pressure and temperature be limited in accordance with the limits specified in the PTLR. The NRC-approval document will be specified in TS 6.9.1.15 and NRC approval will be required in the form of a TS Amendment prior to changing the methodology. Use of P–T limit curves generated using the NRC-approved methods will provide additional protection for the integrity of the reactor vessel, thereby assuring that the reactor vessel is capable of providing its function as a radiological barrier.

TS 3.4.10.3 for Farley Nuclear Plant (FNP) Unit 1 and Unit 2 provides the operability requirements for RCS low temperature overpressure protection (LTOP). Specifically, TS 3.4.10.3 requires that two residual heat removal (RHR) system suction relief valves (RHRRVs) be operable or that the RCS be vented at RCS cold leg temperatures less than or equal to 310[°]F. Consistent with GL 96–03, the Farley Unit 1 and Unit 2 requirements for LTOP will be retained in TS 3.4.10.3 and will be evaluated in accordance with the proposed methodology.

Based on the above evaluation, the proposed changes are administrative in nature and 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.

As stated above, the proposed changes to remove the RCS P–T limits from the TSs and relocate them to the proposed PTLR is an administrative change. Consistent with the guidance provided by GL 96–03, the proposed P–T limits contained in the proposed PTLR meet the requirements of 10 CFR [Part] 50, Appendices G and H, and were generated using the NRC-approved methods described in WCAP–14040–NP–A, Revision 2, as modified at the direction of the NRC Staff. The proposed changes do not result in a physical change to the plant or add any new or different operating requirements on plant systems, structures, or components with the exception of limiting the number of operating RCPs at RCS temperatures below 110[°]F. Limiting the number of operating RCPs below 110[°]F results in a reduction in the [delta]P between the reactor vessel beltline and the RHRRVs, thereby providing additional margin to limits of Appendix G. Provisions are made to allow the start of a second RCP at temperatures below 110[°]F in order to secure the pump that was originally operating without interrupting RCS flow. The LTOP enable temperature exceeds the minimum LTOP enable temperature determined as described in WCAP–14040–NP–A, Rev. 2, thereby providing additional assurance that the LTOP system will be available to protect the RCS in the event of an overpressure transient at RCS temperatures at or below 310[°]F. Based on the methods contained in WCAP–14040–NP–A, Rev. 2, the minimum boltup temperature for the reactor vessel flange region is conservatively established as 70[°]F.

As stated in the above response, implementation of the proposed changes do not result in a significant increase in the probability of a new or different accident (i.e., loss of reactor vessel integrity). The RCS P–T limits will continue to meet the requirements of 10 CFR [Part] 50, Appendices G and H, and will be generated in accordance with the NRC approved methodology described in WCAP–14040–NP–A, Revision 2, as modified at the direction of the NRC Staff. Therefore, the proposed changes do not result in a significant increase in the possibility of a new or different accident from any previously evaluated.

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

The margin of safety is not affected by the removal of the RCS P–T limits from the TSs and relocating them to the proposed PTLR. The RCS P–T limits will continue to meet the requirements of 10 CFR 50, Appendices G and H. To provide additional assurance that the P–T limits continue to meet the requirements of Appendices G and H, TS 6.9.1.15 will require the use of the NRC-approved methodology to generate P–T limits. The RCS LTOP requirements will be retained in TS 3.4.10.3 due to use of the RHRRVs for LTOP, consistent with the guidance provided by GL 96–03, and will be verified to provide adequate protection of the reactor coolant system against the limits of Appendix G. The LTOP enable temperature exceeds the LTOP enable temperature determined in accordance with the NRC-approved methodology, thus protecting the RCS in the event of a low temperature overpressure transient over a broader range of temperatures than required by WCAP–14040–NP–A, Rev. 2. Administrative procedures will preclude operation of the RCS at temperatures below the minimum boltup temperature for the reactor vessel head, thus precluding the possibility of tensioning the reactor vessel head at RCS temperatures below the minimum boltup temperature. Operation of the plant in accordance with the RCS P–T limits specified in the PTLR and continued operation of the LTOP system in accordance with TS 3.4.10.3 will continue to meet the requirements of 10 CFR [Part] 50, Appendices G and H, and will therefore, assure that a margin of safety is not significantly decreased as the result of the proposed changes.

Based on the preceding analysis, SNC [Southern Nuclear Operating Company] has determined that removal of the RCS P–T limits from the TS and relocation to the proposed PTLR will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. SNC therefore concludes that the proposed change meets the requirements of 10 CFR 50.92(c) and 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama.

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

NRC Project Director: Herbert N. Berkow.

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

Date of application request: October 17, 1997.

Description of amendment request: The proposed amendment would modify the technical specifications (TS) for plant heatup and cooldown curves and the maximum allowable power operated relief valve (PORV) setpoint curve for cold overpressure protection, as found in TS Figures 3.4–2, 3.4–3, and 3.4–4. These changes are requested to incorporate information gained from Surveillance Capsule V, which was removed during Callaway Refuel 8 in the fall of 1996 after 9.85 effective full power years (EFPY) of exposure. Capsule V is the third capsule to be removed from the reactor vessel in the continuing surveillance program that monitors the effects of neutron irradiation on the Callway reactor vessel materials under actual plant operating conditions. The proposed changes include:

(1) Figure 3.4–2, heatup limitation curve and Figure 3.4–3, cooldown limitation curve, would be revised to reflect the TR_{NDT} calculated for 20 EFPY in the surveillance capsule report.

(2) Figure 3.4–4 is the maximum allowable PORV setpoint curve for cold overpressure protection. This curve would be (a) revised to account for the changes made in the heatup and cooldown limitation curves, (b) allow for the operation of the normal charging pump, and (c) account for instrument accuracy and other uncertainties.

(3) TS Bases 3/4.4.9 and 3/4.5.2 through 3/4.5.4 would be revised by correcting miscellaneous items and by adding discussion of the normal charging pump.

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.

Pressure and temperature limits for the reactor pressure vessel (RPV) are established to the requirements of 10 CFR 50, Appendix G to ensure brittle fracture of the vessel does not occur. This amendment revises the P/T curves in the TS to reflect the material capsule surveillance results from the sample removed during the fall outage of 1996.

The RPV surveillance capsule contained flux wires for neutron flux monitoring and Charpy V notch impact and tensile test specimens. The irradiated material properties were compared to available unirradiated properties to determine the effect of irradiation on material toughness for the base and weld materials through Charpy testing. Irradiated tensile testing results are compared with unirradiated data to determine the effect of irradiation on the stress-strain relationship of the materials.

The P/T curves are modified to reflect the results of the above examination. These curves and their operating limits were generated using the NRC-approved methods described in WCAP–14040–NP–A, Revision 2 and meet the requirements of 10 CFR 50, Appendices G and H as modified by the provisions of ASME Code Case N–514. The new curves therefore represent the latest information available on the state of the reactor vessel materials. The P/T curves are generated for reactor vessel protection against brittle fracture, they do not affect the recirculation piping. Accordingly, the probability of occurrence of a design basis Loss of Coolant Accident (LOCA) is not increased. Likewise, no other previously evaluated accident and transients, as defined in Chapter 15 of the Final Safety Analysis Report are affected by this proposed change to the Callaway P/T curves. Additionally, this proposed revision does not affect the design, operation, or maintenance of any safety-related system designed for the mitigation or prevention of previously analyzed events.

Since no previously evaluated accidents or transients are affected by this change, their probability of occurrence and consequences is not increased.

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

Implementing the proposed P/T curves into the TS does not alter the design or operation of any system or piece of equipment designed for the prevention or mitigation of accidents and transients. As a result, no new operating modes are introduced from which a new type accident becomes possible. Existing systems will continue to be operated per present design basis assumptions.

The proposed P/T limits were generated from the evaluation of the material capsule removed during the fall outage of 1996 using the NRC-approved methods described in WCAP–14040–NP–A, Revision 2. As a result, these limits include the latest available information on the reactor vessel materials. Furthermore, they will continue to be monitored per the requirements of the TS and 10 CFR 50, Appendices G and H. For the above reasons, the changes do not create the possibility of a new type of accident.

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

The purpose of the P/T limits is to avoid a brittle fracture of the reactor vessel. As such, material capsules are removed periodically to determine the effects of neutron irradiation on the reactor vessel materials. This change to the Callaway curves is proposed to incorporate the evaluation results of the latest capsule removed during the fall outage of 1996. Accordingly, these curves represent the latest information available on the reactor vessel materials.

Also, the curves were generated using the approved methodologies of 10 CFR 50, Appendix G.

The Cold Overpressure Mitigation System Curve (Figure 3.4–4) is also revised to reflect exposure dependencies. This curve was generated for 20 EFPY using approved methodologies and reflects the results of this latest material capsule report. Utilizing the methodology set forth in ASME Section XI, Appendix G, which includes the provisions of Code Case N–514, and 10 CFR 50, Appendices G and H ensures that proper limits and conservative safety factors are maintained.

The proposed changes do not affect the evaluation of any FSAR Chapter 15 transient and accident. Furthermore, the proposed change does not affect the operation of systems or equipment important to safety.

The Limiting Condition for Operation of Specification 3.4.9 will not change. Also, no TS surveillance or surveillance frequencies are revised as a result of this Technical Specification submittal, besides the fact that the P/T surveillance will now refer to the revised curves. Procedures regarding the monitoring of the P/T limits during reactor startup, cooldown, and leakage testing will not change as a result of this proposed Technical Specification change with respect to frequency of the surveillance or the methods used to perform the surveillance. Thus, the P/T limits will continue to be surveilled as before per the same procedures and at the same frequencies.

No other Technical Specifications are affected by the proposed revision. The margin of safety to any Technical Specification safety limit therefore is not reduced. For the above reasons the new curves do not represent a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

Date of application request: October 31, 1997.

Description of amendment request: The proposed amendment would revise the engineered safety features actuation system (ESFAS) Functional Unit 6.f, Loss of Offsite Power-Start Turbine-Driven Pump, in Technical Specification Tables 3.3–3, 3.3–4, and 4.3–2. The tables would be revised to create separate functional units for the analog and digital portions of the ESFAS function associated with starting the turbine-driven auxiliary feedwater pump (TDAFP) upon a loss of offsite power.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The recognition that different operability and surveillance requirements apply to analog vs. digital circuitry does not impact any previously analyzed accidents. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change does not alter the current method or procedures for meeting the surveillance requirements in Table 4.3–2. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The separation of analog and digital portions of Functional Unit 6.f will not impact the normal method of plant operation.

The operability requirements, ACTION Statement, and surveillance requirements for the analog portion, new Functional Unit 6.f.1), are identical to those of Functional Unit 8.a, while the requirements for the digital portion, new Functional Unit 6.f.2), are consistent with the current Technical Specifications, other than the new ACTION Statement 39 provisions that defer to the TDAFP Specification 3.7.1.2 requirements and the performance of a TADOT during appropriate plant conditions. These changes do not change any ESFAS design standards and are appropriate for digital functions such as this. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

Date of application request: October 31, 1997.

Description of amendment request: The proposed amendment would change Technical Specification Tables 2.2–1, 4.3–1, and 3.3–4, as well as their associated Bases, in order to reduce repeated alarms, rod blocks, and partial reactor trips that continue to manifest themselves, especially during beginning of cycle operation following refueling outages. Besides the potential for distracting operator attention away from more safety significant evolutions, these occurrences have also led to power reductions during surveillance testing in order to avoid reactor trips, since the channel being tested is placed in the tripped condition. These changes to various setpoint terms associated with the overtemperature delta T, overpower delta T, and steam generator (SG) water level low-low vessel delta T (Power-1 and Power-2) reactor trip and auxiliary feedwater (AFW) start engineered safety feature actuation system (ESFAS) functions will improve plant operations and reduce the potential for unnecessary reactor trips, with no detrimental effect on the plant's safety analysis or licensing basis.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed changes will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed changes will not affect the probability of any event initiators nor will the proposed changes affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unchanged. The Overtemperature delta T Allowable Value increase is justified by the use of existing setpoint margin and elimination of conservatisms not required by the safety analysis and licensing basis. There will be a reduction in the incidence of alarms, rod stops, and partial reactor trips. There will also be less of a need to reduce power during on-line surveillance testing. These changes represent substantial plant operational improvements.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the 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 a margin of safety.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). Maintaining the SAL preserves the margin of safety.

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

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

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

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

NRC Project Director: William H. Bateman.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of amendment request: December 4, 1997.

Description of amendment request: This amendment as reflected in Section 2.1.1.2 of the Technical Specifications would continue the use of the existing Siemens Power Corporation minimum critical power ratio (MCPR) safety limits for Cycle 14 and would change the Asea Brown Boveri (ABB) MCPR safety limit for single loop operation from 1.08 for Cycle 13 to 1.09 for Cycle 14.

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 proposed change 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. The proposed Technical Specifications amendment continues the use of conservatively established ATRIUM–9X MCPR safety limits for WNP–2 such that the fuel is protected during normal operation as well as during plant transients or anticipated operational occurrences.

The probability of an evaluated accident is not increased by the continued use of the interim ATRIUM–9X MCPR safety limit of 1.13 (two loop operation) or 1.14 (single loop operation) or from changing the ABB single loop MCPR safety limit of 1.08 (Cycle 13) to 1.09 (Cycle 14). The changes do not require any physical plant modifications, physically affect any plant component, or entail changes in plant operation. The increase in single loop MCPR safety limit is attributed to a slightly more conservative assembly power distribution used in the Cycle 14 calculations following ABB standard methodology. While the Cycle 13 result is also conservative, the increase in Cycle 14 is intended to accommodate small cycle to cycle variability. Therefore, no individual precursors of an accident are affected.

This Technical Specification amendment proposes to continue using the interim MCPR safety limits for ATRIUM–9X fuel to protect the fuel during normal operation as well as during plant transients or anticipated operational occurrences. The method that is used to determine the ATRIUM–9X additive constant uncertainty is conservative, such that the resulting interim ATRIUM–9X MCPR safety limits are high enough to ensure that less than 0.1% of the fuel rods are expected to experience boiling transition if the limit is not violated. Using NRC approved methodology, ABB has utilized these interim values as the basis for the Cycle 14 safety limit for the co-resident ATRIUM–9X. Operational limits have been established based on the interim ATRIUM–9X MCPR safety limits to ensure that the safety limits are not violated. This will ensure that the fuel design safety criteria (more than 99.9% of the fuel rods avoid transition boiling during normal operation as well as anticipated operational occurrences) is met. In addition, since the operability of plant systems designed to mitigate any consequences of accidents have not changed, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications of the plant configuration or allowable modes of operation. This Technical Specification change continues the use of added conservatism in the ATRIUM–9X MCPR safety limits which resulted from analytical changes and use of an expended database. Also, ABB has calculated single loop MCPR safety limit [which is] about 0.006 greater in Cycle 14 than was used in Cycle 13. The increase in single loop MCPR safety limit is attributed to a slightly more conservative assembly power distribution used in the Cycle 14 calculations following ABB standard methodology. While the Cycle 13 result is also conservative, the increase in Cycle 14 is intended to accommodate small cycle to cycle variability. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

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

The continued use of interim MCPR safety limits provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated. These interim limits are based on calculations by SPC using the revised ATRIUM–9X additive constant uncertainty. These calculations are based on a larger pool of data than previous calculations (527 data points versus 82 data points). Additionally, the revised additive constant uncertainty has been conservatively applied in the calculation of the interim ATRIUM–9X MCPR safety limits resulting in more restrictive limits.

The calculated single loop MCPR safety limit results are about 0.006 greater for Cycle 14 than they were for Cycle 13. The increase in single loop MCPR safety limits is attributed to a slightly more conservative assembly power distribution used in the Cycle 14 calculations following ABB standard methodology. Because the fuel design safety criteria of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as anticipated operational occurrences is met, there is not a significant reduction in the margin of safety.

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

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

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

NRC Project Director: William H. Bateman.

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.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: December 11, 1997.

Brief description of amendment request: The proposed amendment would provide a one-time change to the Technical Specifications to allow purging of the containment during Modes 3 (Hot Standby) and 4 (Hot Shutdown) upon return to power from the current refueling outage (1R13).

Date of publication of individual notice in Federal Register: December 18, 1997 (62 FR 66397).

Expiration date of individual notice: January 20, 1998.

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

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.

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

Date of application for amendments: May 6, 1997, as supplemented on July 30, 1997.

Brief description of amendments: The amendments will change Technical Specification 3/4.7.5, ''Ultimate Heat Sink'' and the associated Bases to support steam generator replacement and incorporate recent Ultimate Heat Sink (UHS) design evaluations.

Date of issuance: December 12, 1997.

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

Amendment Nos.: 95 and 95.

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

Date of initial notice in **Federal Register**: July 2, 1997 (62 FR 35847). The July 30, 1997, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

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

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

Date of application for amendments: September 30, 1997.

Brief description of amendments: The amendments would add a new Technical Specification (TS) Section 3/4.12.C and associated bases to allow certain reactor coolant pressure tests to be performed in MODE 4 when the reactor pressure vessel requires testing at temperatures greater than 212 degrees Fahrenheit. This temperature normally corresponds with MODE 3.

Date of issuance: January 5, 1998.

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

Amendment Nos.: 164, 159, 179 and 177.

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

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61839). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 5, 1998.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: April 24, 1997.

Brief description of amendment: The amendment revises the inservice inspection requirements associated with steam generator tube sleeves.

Date of issuance: December 23, 1997.

Effective date: December 23, 1997.

Amendment No.: 187.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 16, 1997 (62 FR 38134).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 15, 1997.

Brief description of amendment: The amendment revises the license to reflect the transfer of the 30-percent undivided ownership interest in the River Bend Station, Unit No. 1 from Cajun Electric Power Cooperative, Inc. to Entergy Gulf States, Inc. The transfer was approved by Order dated November 28, 1997, which was published in the **Federal Register** on December 5, 1997 (62 FR 64404).

Date of issuance: December 23, 1997.

Effective date: December 23, 1997.

Amendment No.: 101.

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

Date of initial notice in **Federal Register**: October 24, 1997 (62 FR 55432).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1997.

No significant hazards consideration comments received: No.

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

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: July 18, 1997, as supplemented September 12 and October 25, 1997.

Brief description of amendment: Establish a new Low-Temperature Overpressure Protection Technical Specification (TS).

Date of issuance: December 22, 1997.

Effective date: December 22, 1997.

Amendment No.: 161.

Facility Operating License No. DPR–31: Amendment revised the TSs.

Date of initial notice in **Federal Register**: August 13, 1997 (62 FR 43369) The supplemental letters dated September 12 and October 25, 1997 did not change the initial no significant hazards consideration determination or expand the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1997.

No significant hazards consideration comments received: No.
Local Public Document Room location: Coastal Region Library, 8619 W. Crystal River, Florida 34428.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: July 29, 1997, as supplemented October 29, 1997.
Brief description of amendment: The amendment revises the Post-Accident Monitoring Instrumentation Technical Specification (TS).
Date of issuance: December 22, 1997.
Effective date: December 22, 1997.
Amendment No.: 162.
Facility Operating License No. DPR–31: Amendment revised the TSs.
Date of initial notice in **Federal Register**: August 13, 1997 (62 FR 43369).
The supplemental letter October 29, 1997 did not change the initial no significant hazards consideration determination.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1997.
No significant hazards consideration comments received: No.
Local Public Document Room location: Coastal Region Library, 8619 W. Crystal River, Florida 34428.

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

Date of application for amendments: February 24, 1997, as supplemented on April 24 and December 4, 1997.
Brief description of amendments: The admendments change technical specification section 6.9.1.7, Core Operating Limits Report, to reflect use of the Westinghouse Best Estimate Large Break Loss-of-Coolant Accident (LOCA) methodology for large break LOCA analysis, including supporting documents.
Date of issuance: December 20, 1997.
Effective date: December 20, 1997.
Amendment Nos.: 195 and 189.
Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the TS.
Date of initial notice in **Federal Register**: June 4, 1997 (62 FR 30631). By letter dated December 4, 1997, the licensee provided additional information which did not affect the original no significant hazards determination.
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 1997.
No significant hazards consideration comments received: No.
Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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: October 7, 1997, as supplemented December 17, 1997.
Brief description of amendment: Technical Specifications 4.6.1.1, 3/4.6.1.2, and 3/4.6.1.3 require the testing of the containment to verify leakage limits at a specified test pressure. The amendment (1) modifies the list of valves that can be opened in Modes 1 through 4, (2) removes a footnote on Type A testing, and (3) makes editorial changes to the Technical Specifications and makes changes to the associated Bases sections.
Date of issuance: December 18, 1997.
Effective date: As of the date of issuance, to be implemented within 60 days.
Amendment No.: 154.
Facility Operating License No. NPF–49: Amendment revised the Technical Specifications. Date of initial notice in **Federal Register**:
November 5, 1997 (62 FR 59917) The December 17, 1997, letter provide clarifying information that did not change the October 7, 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 December 18, 1997.
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 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of application for amendment: October 6, 1997.
Brief description of amendment: The requested changes would increase the allowable band for control and shutdown rod demanded position versus indicated position from plus or minus 12 steps to plus or minus 18 steps when the power level is not greater than 85% rated thermal power. The changes have already been approved for Salem Unit 2 in Amendment No. 183, issued September 10, 1997, as an exigent amendment.
Date of issuance: December 22, 1997.
Effective date: December 22, 1997.
Amendment No.: 202.
Facility Operating License No. DPR–70: This amendment revised the Technical Specifications.
Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61845).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1997.
No significant hazards consideration comments received: No.
Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

Date of application for amendment: June 19, 1997, with additional information provided on August 15, 1997 (TS 391T).
Brief Description of amendment: The amendments revise the Technical Specifications (TS) to temporarily extend the allowed outage time for the emergency diesel generators from 7 to 14 days to permit completion of preventive maintenance.
Date of issuance: December 22, 1997.
Effective Date: December 22, 1997.
Amendment Nos.: 250 and 209.
Facility Operating License Nos. DPR–52 and DPR–68: Amendment revised the TS.
Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40858).
The additional information provided on August 15, 1997 does not affect the staff's proposed finding of no significant hazards consideration.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1997.
No significant hazards consideration comments received: None.
Local Public Document Room Location: Athens Public library, 405 E. South Street, Athens, Alabama 35611.

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

Date of amendment request: July 9, 1996 (TXXX–96393), as supplemented on December 12, 1997 (TXXX–97268). (The supplement contains clarifying information and does not change the staff's original proposed no significant hazards determination.)

Brief description of amendments: The amendments change Technical Specification 3.3–3, ''Engineered Safety Features Actuation System Instrumentation Trip Setpoints.'' The proposed changes would increase the minimum allowable value of the Unit 1 Steam Line Pressure—Low Safety Injection and Steam Line Isolation functions. These changes are needed to ensure that the instrumentation error is properly accounted for in the TSs.

Date of issuance: December 30, 1997.

Effective date: December 30, 1997, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 56; Unit 2—Amendment No. 42.

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

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6579) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 30, 1997.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 7th day of January 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–753 Filed 1–13–98; 8:45 am]

BILLING CODE 7590–13–P

OFFICE OF MANAGEMENT AND BUDGET

Proposed Revision of OMB Circular A–97

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Proposed revision of OMB Circular A–97.

SUMMARY: The Office of Management and Budget requests agency and public comments on a proposed revision of OMB Circular No. A–97, ''Rules and Regulations Permitting Federal Agencies to Provide Specialized or Technical Services to State and Local Units of Government, Under Title III of the Intergovernmental Cooperation Act of 1968.'' The proposed revision establishes and updates Circular A–97 requirements with regard to the provision or receipt of commercial support services to or from Federal agencies and State and local governments. Circular A–97 was issued on April 29, 1969, and was last revised on March 27, 1981.

DATES: Agency and public comments are due to the Office of Management and Budget (OMB) not later than March 16, 1998.

ADDRESSES: Written comments should be sent to the Budget Analysis and Systems Division, NEOB Room 6002, Office of Management and Budget, 725 17th Street, N.W., Washington, D.C. 20503. FAX terminal comments may be sent to (202) 395–7230.

AVAILABILITY: Copies of Circular A–97 may be obtained by contacting The Executive Office of the President, Office of Administration, Publications Office, Washington, D.C. 20503, at (202) 395–7332.

FOR FURTHER INFORMATION CONTACT: The Budget Analysis and Systems Division, NEOB Room 6002, Office of Management and Budget, 725 17th Street, N.W., Washington, D.C. 20503, Telephone Number: (202) 395–6104, FAX Number (202) 395–7230.

BACKGROUND: Circular A–97 has become an integral part of the Federal privatization and outsourcing discussion. Federal support to meet State and local workload requirements has been suggested for a wide range of commercial services, including payroll services, background investigations services, leasing management, fleet management, geodetic and mapping services, prison requirements and health care services. Economies of scale, similarities of purpose and approach, and the possibility of a partnership to meet common data requirements suggest there may be opportunities for Federal or State and local taxpayer savings. On the other hand, special care must be taken to ensure that the Federal Government does not, unnecessarily, become a reimbursable competitor with or otherwise displace private sector, State or local employees. To address these concerns, OMB has prepared a revised and updated Circular A–97. OMB requests comments on this revision.

Franklin D. Raines,

Director.

To the Heads of Executive Departments and Establishments

Subject: Rules and regulations permitting Federal agencies to provide specialized or technical services to State and local units of government under Title III of the Intergovernmental Cooperation Act of 1968

1. Purpose

This Circular promulgates the rules and regulations that the Director of the Office of Management and Budget (OMB) is authorized to issue pursuant to Section 302 of the Intergovernmental Cooperation Act of 1968 (Pub. L. 90–577; 82 Stat. 1102). It also provides for the coordination of the action of Federal departments and agencies (hereinafter referred to as ''Federal agencies'') in exercising the authority contained in Title III of said Act as directed by the President's Memorandum of November 8, 1968 (33 FR 16487).

2. Background

a. Title III of the Intergovernmental Cooperation Act of 1968 is intended to:

1. Encourage intergovernmental cooperation in the conduct of specialized or technical services and provision of facilities essential to the administration of State or local governmental activities.

2. Enable State and local governments to avoid unnecessary duplication of special service functions.

3. Authorize Federal agencies that do not have such authority to provide reimbursable specialized and technical services to State and local governments.

b. Title III of the Act authorizes the head of any Federal agency, upon a written request from a State or political subdivision thereof, to provide specialized or technical services, upon payment to the Federal agency by the unit of government making the request, of salaries and all other identifiable direct and indirect costs of performing such services. These costs shall be established in accordance with all applicable statements of Federal financial accounting standards.

c. Title III of the Act requires that:

1. Any services provided pursuant to Title III shall include only those that the Director of the Office of Management and Budget through rules and regulations determines Federal agencies have special or unique competence to provide.

2. The Director's rules and regulations shall be consistent with, and in furtherance of, the Government's policy of relying on the private enterprise system to provide those services that are reasonably and expeditiously available through ordinary business channels.

3. All moneys received by any Federal agency in payment of furnishing specialized and technical services under