views may be presented by the members of the public, including members of the nuclear industry. Persons desiring to make oral statements should notify Mr. Frank P. Gillespie (Telephone 301/415–1004, e-mail FPG@nrc.gov) or Mr. Mohan C. Thadani (Telephone 301/415–1476, e-mail MCT@nrc.gov) five days prior to the meeting date, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras will be permitted during this meeting.

Further information regarding topics of discussion; whether the meeting has been canceled, rescheduled, or relocated; and the Panel Chairman's ruling regarding requests to present oral statements and time allotted, may be obtained by contacting Mr. Frank P. Gillespie or Mr. Mohan C. Thadani between 8:00 a.m. and 4:30 p.m. EDT.

PPEP meeting transcripts and meeting reports will be available from the Commission's Public Document Room. Transcripts will be placed on the agency's web page when a web site for PPEP is established.

Dated: August 5, 1999.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 99–20656 Filed 8–10–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of August 9, 16, 23, and 30, 1999.

PLACE: Commissioner's Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of August 9

Thursday, August 12

11:30 a.m. Affirmation Session (Public Meeting) (If needed).

Week of August 16—Tentative

There are no meetings scheduled for the Week of August 16.

Week of August—Tentative

Tuesday, August 24

2:00 p.m. Briefing by Executive Branch (Closed—ex. 1)

3:30 p.m. Briefing on Threat Assessment (Closed—ex. 1)

Wednesday, August 25

9:55 a.m. Affirmation Session (Public Meeting) (If needed)

Week of August 30—Tentative

Wednesday, September 1

9:25 a.m. Affirmation Session (Public Meeting) (If needed)

2:00 Briefing on PRA Implementation Plan (Public Meeting) (Contact: Tom King, 301–415–5790)

* The schedule for Commission meeting is subject to change on short notice. To verify the status of meetings call (recording)—(301)–415–1292. Contact person for more information: Bill Hill (301) 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: August 6, 1999.

William M. Hill, Jr.,

SECY, Tracking Officer, Office of the Secretary.

[FR Doc. 99–20906 Filed 8–9–99; 8:45 am]

BILLING CODE 7590–01–M

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 July 17, 1999, through July 30, 1999. The last biweekly notice was published on July 28, 1999 (64 FR 40903).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–

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

By September 10, 1999, 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.

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

Date of amendment request: July 9, 1999.

Description of amendment request: The proposed amendment would revise Harris Nuclear Plant (HNP) Technical Specification (TS) 3/4.2.2, ''Heat Flux Hot Channel Factor—$F_Q(Z)$,'' TS 3/4.2.3, ''RCS Flow Rate And Nuclear Enthalpy Rise Hot Channel Factor,'' TS 3/4.2.5, ''DNB Parameters,'' an associated note in TS Table 2.2–1, and associated Bases. Specifically, the proposed amendment would: (1) Remove the allowance for reduced power operation for reduced Reactor Coolant System (RCS) flow rate conditions; (2) separate the requirements for F delta H and RCS flow rate in the format prescribed by NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995; and, (3) implement the guidance of NUREG–1431, Revision 1, and NRC Generic Letter (GL) 88–16, dated October 4, 1988 for TS 3/4.2.2, TS 3/4.2.3, TS 3/4.2.5 and associated Bases by removing cycle specific parameters and placing that information into the Core Operating Limits Report (COLR).

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

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

The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS.

As described in HNP TS Bases, the limits on heat flux hot channel factor, RCS flow rate, and enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR [departure from nucleate boiling ratio] are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200 degree Fahrenheit ECCS [emergency core cooling system] acceptance limit.

Removing the allowance for reduced power operation for reduced RCS flow conditions is more restrictive than that currently allowed by TS. Power Distribution Limiting Conditions for Operation for heat flux hot channel factor and enthalpy rise hot channel factor are not affected by this change. Therefore, the consequences of an accident will not increase because of this change. Power Distribution limits place administrative restrictions on reactor core parameters and as such do not initiate nor mitigate accidents.

Power Distribution limits at HNP are developed using NRC approved methodologies. Changing power distribution limits to be consistent with NUREG–1431, Revision 1 will not increase the probability or consequences of an accident that has been previously evaluated.

Relocating cycle specific information from TS to the COLR will not impact the ability of structures, systems, or components to mitigate accidents. Future changes to relocated requirements in the COLR will be submitted to the NRC for review in accordance with HNP TS Section 6.

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

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

The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS. The changes are consistent with NUREG–1431, Revision 1 and the Commission's Final Policy Statement on Technical Specification improvements. The proposed amendment will not create any new accident scenarios, because the change does not introduce any new single failures, adverse equipment or material interactions, or release paths.

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

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

The LCO limit for RCS flow rate at 100.0% reactor power has not changed. The previous capability to operate with reduced RCS flow rate has been eliminated. This aspect of the proposed change is more restrictive than current plant TS in that continued reactor operation greater than 5% is not allowed if RCS flow rate is less than the LCO limit at 100% power.

Changes to TS 3/4.2.2, TS 3/4.2.3, TS 3/4.2.5 and associated Bases are in accordance with NUREG–1431, Revision 1. The completion times for TS Actions are acceptable because the plant is not allowed to remain in an unacceptable condition for an extended period of time. Sufficient time to reduce reactor power in an orderly manner or perform other required actions is also provided. The surveillance intervals established by NUREG–1431, Revision 1 have been determined to be adequate for monitoring the change in power distribution.

Relocating cycle specific information from HNP TS to the COLR is in accordance with NRC GL 88–16. HNP does not intend to alter the methodologies for any parameter limit calculation as a result of this change. The proposed change is in accordance with the plant safety analysis. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 9, 1999.

Description of amendment request: The proposed amendment would relocate Harris Nuclear Plant (HNP) Technical Specification (TS) 3/4.3.3.3, ''Seismic Instrumentation,'' TS 3/4.3.3.4, ''Meteorological Instrumentation,'' TS 3/4.3.3.9, ''Metal Impact Monitoring System,'' and TS 3/4.3.3.11, ''Explosive Gas Monitoring Instrumentation,'' to plant procedure PLP–114, ''Relocated Technical Specifications and Design Basis Requirements.'' The proposed change is in accordance with guidance provided by NRC Generic Letter 95–10, ''Relocation of Selected Technical Specification Requirements Related to Instrumentation.'' Changes to relocated requirements would be performed in accordance with 10 CFR 50.59. *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation are not accident initiating components as described in the Final Safety Analysis Report. Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation are not accident mitigating components. There are no modifications being made to plant systems as a result of this change. Additionally, there are no changes being made to the way in which systems are being operated as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation are not accident initiating components as described in the Final Safety Analysis Report (FSAR). The proposed change relocates the TS requirements for Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation to plant procedure PLP–114. Plant systems and components are not modified as a result of this change. Future changes in these systems will be controlled in accordance with 10 CFR 50.59.

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

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

The proposed change to Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation does not affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or the FSAR. Accordingly, NRC Acceptance Limits are not affected by this change. The proposed change relocates the TS requirements for Seismic Instrumentation, Meteorological Instrumentation, Metal Impact Monitoring System, and Explosive Gas Monitoring Instrumentation to plant procedure PLP–114. Plant systems and components are not modified as a result of this change. Future changes in these systems will be controlled in accordance with 10 CFR 50.59. Generic Letter 95–10 states that the staff has concluded that these provisions are not related to dominant contributors to plant risk.

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

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

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

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

NRC Section Chief: Sheri R. Peterson.

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

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

Date of amendment request: June 30, 1999

Description of amendment request: The proposed amendment would clarify that the source of DC electrical power required for a unit in Mode 5 or 6 or during the movement of irradiated fuel assemblies may be cross-tied to the opposite unit. An administrative change would also delete reference to AT&T batteries since all AT&T batteries have been replaced with Charter Power Systems, Inc. (C&D) batteries. The amendment would also remove the Allowed Outage Time (AOT) extension approved for Braidwood Station by Amendment No. 99. The activity addressed by Amendment No. 99 is complete and the extension no longer applies.

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 proposed change will allow one DC bus on a shutdown unit to be supplied via the DC bus cross-tie to the opposite unit. The other DC bus on the shutdown unit will at all times be required to be fully operable, supplied by the associated battery and charger, and the associated cross-ties open. The DC electrical system is not considered an initiator of any accident previously evaluated, and therefore the probability of a previously analyzed accident is unchanged.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operated in response to the analyzed event, and the setpoints at which these actions are initiated. Sufficient equipment remains available to mitigate the consequences of previously analyzed events. The Updated Final Safety Analysis Report (UFSAR) section 8.3.2.1.1 clearly allows operation with the DC cross-tie closed on one DC bus between a unit that is operating and a unit that is shutdown, or between two shutdown units, in the manner proposed by this amendment. The TS in effect prior to the implementation of the Improved TS also allowed operation in the manner proposed by this amendment. If DC buses are cross-tied due to an inoperable DC source on a shutdown unit, both the previous TS and the change proposed by this amendment limit the time in this condition to seven days, and if the inoperable source is a battery, the current on the cross-tie is limited to 200 amps. These actions protect both the operating unit, and the shutdown unit. If a shutdown unit's DC bus is cross-tied to an operating unit's DC bus due to an inoperable charger on the operating unit, both the previous TS and the change proposed by this amendment limit the time in this condition to 24 hours. The limitations imposed by both the previous TS and the change proposed by this amendment ensure that operation in this configuration is within the design bases of the plant. Thus the consequences of accidents previously analyzed are unchanged between the previous TS and the change proposed by this amendment. In the worst case scenario, assuming a single failure, one DC bus on the shutdown unit will always be operable, and the ability to mitigate the consequences of any accident previously analyzed is preserved.

The change to delete all references in the Braidwood TS to AT&T batteries and the AOT extension granted under TS Amendment Number 99 is administrative only, and has no impact on the probability or consequences of accidents previously evaluated.

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

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

The proposed change does not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated. There are no setpoints affected by this change at which protective or mitigative actions are initiated. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits in being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change to delete all references in the Braidwood TS to AT&T batteries and the AOT extension granted under TS Amendment Number 99 is administrative only, and cannot create the possibility of a new or different kind of accident.

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change, which will allow one DC bus on a shutdown unit to be supplied via the DC bus cross-tie to the opposite unit, is acceptable because of the limitations imposed on operation in this configuration, and because the other DC bus on the shutdown unit will at all times be required to be fully operable, supplied by the associated battery and charger, and the associated cross-ties open. The TS in effect prior to the implementation of the Improved TS allowed operation in the manner proposed by this amendment. In the worst case scenario, assuming a single failure, one DC bus on the shutdown unit will always be operable. Thus, there is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not reduce the margin of safety.

The change to delete all references in the Braidwood TS to AT&T batteries and the AOT extension granted under TS Amendment Number 99 is administrative only, and does not 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: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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: May 3, 1999.

Description of amendment request: The proposed amendments would relocate Technical Specifications (TS) Section 3/4.6.I to the Updated Final Safety Analysis Report (UFSAR). TS Section 3/4.6.I contains reactor coolant chemistry limiting conditions for operation (LCO) and surveillance requirements (SR) for conductivity, chloride concentration, and pH.

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:

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

The proposed changes simplify the TS, meet regulatory requirements for relocated TS's, and implement the recommendations of the NRC Final Policy Statement on TS improvements. The Chemistry requirements will be relocated to the Updated Final Safety Analysis Report (UFSAR) and to applicable station procedures. Future changes to these requirements will be controlled by 10 CFR 50.59. The proposed changes are administrative in nature and do not involve any modification to any plant equipment or affect plant operation. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any previously evaluated accident.

Consequently, this proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety related system performs its function. Therefore, this proposed TS amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment represents the relocation of current requirements, which are based on generic guidance or previously approved provisions for other stations. The proposed changes are administrative in nature 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 Nuclear Power Station. Since the proposed changes are administrative in nature, and are based on NRC accepted provisions which have been adopted at other nuclear facilities, and maintain the necessary levels of system reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: July 7, 1999.

Description of amendment request: The proposed amendments would (1) revise Technical Specification Section 2.1, Safety Limits, to reflect a change to the LaSalle, Unit 1, Minimum Critical Power Ratio Safety Limit; and (2) revise Technical Specification Section 6.6.A.6 to add an NRC-approved Siemens Power Corporation methodology to the list of topical reports used to determine the core operating 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. Does the 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. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

Changing the MCPR Safety Limit for LaSalle Unit 1 will not increase the probability or the consequences of an accident previously evaluated. This change implements the MCPR Safety Limit resulting from the SPC ANFB critical power correlation methodology using the approved ATRIUM–9B additive constant uncertainty. For each cycle, cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. The MCPR Safety Limit ensures that less than 0.1% of the rods in the core are expected to experience boiling transition. Therefore the probability or consequences of an accident will not increase.

Adding EMF–85–74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A) to Section 6 does not increase the probability or consequences of an accident previously evaluated. The NRC-approved burnup extension for RODEX2A applications has been demonstrated to meet all applicable design criteria. Therefore adding this methodology to Technical Specification Section 6 does not increase the probability or consequences of an accident previously evaluated .

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

2. Does the 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 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.

Changing the MCPR Safety Limit does not create the possibility of a new accident from any accident previously evaluated. This change does 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 LaSalle Unit 1 to support Cycle 9 operation. This change does not introduce any physical changes to the plant, alter the processes used to operate the plant, or change allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

The addition of RODEX2A (EMF–85–74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A)) does not create the possibility of a new accident from an accident previously evaluated. This change does not alter or add any new equipment or change modes of operation. This change does not introduce any physical changes to the plant, alter the processes used to operate the plant, or change allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

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

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

Changing the MCPR Safety Limit for LaSalle Unit 1 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 calculated to be in boiling transition. The proposed Technical Specification amendment request reflects the MCPR Safety Limit results from evaluations by SPC using NRC-approved methodology.

The revised MCPR Safety Limit will ensure the same 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[al] 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.

The addition of EMF–85–74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A) to Section 6 does not decrease the margin of safety. The burnup limit extension for RODEX2A applications has been reviewed and approved by the NRC. The data supporting the burnup extension demonstrates that all applicable design criteria are met. Therefore, since the burnup extension is acceptable and within the design criteria, using the approved burnup extension will not affect the margin of safety.

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

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 9, 1998, as supplemented on July 7, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification Table 3.3.3–2, ''Emergency Core Cooling System Actuation Instrumentation Setpoints,'' to modify the degraded voltage second level undervoltage relay setpoint and allowable value. These proposed amendments were originally noticed on January 13, 1999 (64 FR 2245), and are being renoticed to include the revised setpoints that were included in the July 7, 1999, supplement.

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

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

The setpoint change does not change the logic or function of the degraded voltage protection circuits as described in the UFSAR [Updated Final Safety Analysis Report] Section 8.2.3. They also do not reduce the reliability of these circuits. The increase in the degraded voltage protection circuit setpoint is conservative compared to the existing setpoint. There is no change as a result of this amendment to the underlying accident and transient analyses that support operations of LaSalle County Station. Inadvertent or spurious operation of the degraded voltage protection function will initiate loading of the safe shutdown loads on the diesel generators and is not assumed to initiate an accident. The proposed degraded voltage setpoints are low enough to prevent spurious actuations given the expected offsite grid voltages. After implementation of this amendment, no operator actions are required for equipment operations in response to degraded voltage conditions.

This change does not affect the initiators or precursors of any accident previously evaluated. This change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure.

The consequences of accidents previously evaluated are not increased. The proposed change does not affect the required level of availability of systems required to mitigate the accidents considered in the analyses. The proposed changes will ensure that the Class 1E equipment will be capable of starting and operating during a design basis accident with degraded offsite grid voltage. The increase in the level of confidence is the result of more rigorous methodology used to determine limiting Class 1E bus voltages at the minimum expected offsite AC voltage. These calculations demonstrate that the degraded voltage relays will not actuate following a block start of the electrical loads that are automatically actuated by or as a consequence of the LOCA [loss-of-coolant accident] signal if the switchyard voltage remains above 352 kV.

If the grid voltage drops below 352 kV, then the analytical limit of 3814 volts for proper operation of class 1E loads connected to each 4.16 kV Class 1E bus is assured by transfer to the respective onsite power sources (Emergency Diesel Generators (EDGs)) by the degraded voltage logic.

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

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

Setpoint methodology established the bases to ensure that, with known errors, the relays will detect degraded voltage conditions and transfer safety loads to the EDGs at a voltage level adequate to ensure proper safety equipment performance and to prevent equipment damage.

The trip setpoint of greater than or equal to 3863 volts and less than or equal to 3877 volts and the allowable value of greater than or equal to 3814 volts and less than or equal to 3900 volts, include adequate tolerance to calibrate the relay trip units while ensuring that the Class 1E bus voltage will remain above the analytical limits.

These setpoint changes will ensure that adequate voltages will be available for the continuous operation of safety-related equipment required to function during a LOCA. These proposed changes will also ensure that adequate voltages will be available for starting any Class 1E equipment.

The proposed degraded voltage setpoint change does not change the design of the degraded voltage protection system or its function to protect against degraded offsite power. Actuation of the degraded voltage protection system will initiate a sequence of events that will start the EDG for the associated Class 1E bus, strip loads from the Class 1E bus, open all feed breakers to the Class 1E bus, close the Emergency feed breaker (thus energizing the Class 1E bus from the respective EDG), and initiate starting of the Safe Shutdown equipment supplied by the Class 1E bus.

Since the scope of this change does not affect the operation of auxiliary power system or any actions necessary to mitigate the consequences of accidents or achieve safe shutdown, the change does not involve a new or different accident scenario.

Therefore, these 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 the margin of safety because:

The proposed amendment will allow the degraded voltage setpoint to be conservatively established based on new engineering calculations which consider the lowest expected offsite grid voltage and operation of required Class 1E equipment under design basis accident loading conditions.

The proposed degraded voltage setpoints will ensure that adequate Class 1E bus voltage will be available to support starting and operation of required Class 1E loads. The proposed setpoint includes instrument error to ensure that the lowest possible voltage will not be lower than the degraded voltage analytical limits. Additionally, the proposed setpoints are low enough to prevent spurious actuations due to expected fluctuations in the grid voltage. The new setpoints are also set with margin to the minimum Class 1E bus voltage, which is based on a minimum grid voltage of 352 kV, which is less than the expected grid voltage of 354 kV. The proposed changes will provide an increase in the level of protection that currently exists and will ensure the margin of safety is adequately maintained.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: April 20, 1999 (Reference NRC–99–0035).

Description of amendment request: The proposed amendment will revise the Technical Specifications by deleting Specification D.3.c. Specification D.3.c requires the licensee to perform weekly observations of the nitrogen cover gas pressure within the sodium storage tanks located in the Sodium Building Complex. Removing this surveillance requirement would allow the licensee to remove the nitrogen cover gas system from service for these sodium storage tanks. This action is necessary for the licensee to begin work on removing the remaining residual sodium from these tanks. The licensee also requested an editorial change to delete the words ''STORAGE TANK'' from the title of Specification D.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

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

Removing the primary cover gas supply from the storage tanks will not significantly increase the probability of an accident occurring as long as the probability of an uncontrolled water reaction with residual sodium is not significantly increased. This is ensured by sealing the storage tanks after the nitrogen cover gas system is removed except when controlled activities such as sampling are performed. The consequences of an accident would not be affected by removing the nitrogen cover gas supply from service as the previously analyzed primary sodium accident already involves release of all the radioactive material in the primary sodium. Removing the cover gas will not increase the amount of radioactive material available to be released.

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

A sodium accident has been previously evaluated. No other type of accident could be caused by removing the primary sodium tanks cover gas or opening the tanks since no other system or mode of operation of any other system will be affected.

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

Currently, only a small amount of residual sodium remains in the primary sodium storage tanks. Some of this residual sodium may have been converted to sodium carbonate. This conversion of sodium to sodium carbonate would have left even less sodium remaining in these tanks. The cover gas is a good precaution, especially for tanks sitting unattended for many years. It prevents moisture from intruding into the tanks and reacting with the sodium residues. It also prevents oxygen from entering these tanks and reacting with any hydrogen formed from reactions of water and sodium. Discontinuing the use of cover gas slightly reduces the margin of safety, but not significantly. Removing the cover gas does not, in itself, introduce water into the tank in an uncontrolled manner. Even if slight amounts of moisture from humidity in the air enter these tanks over the next year or two, until the sodium is removed while the tanks are either opened or sealed, the volume of each tank (15,000 gallons) is large enough that the tank should be able to dissipate any small reactions that could occur. The design pressure for the primary sodium storage tanks is from vacuum to 50 pounds per square inch based on the vendor's drawing.

Even if sufficient water entered the tank, generated hydrogen, and sufficient oxygen entered the tank to cause a reaction that released the contents of the tank, there would be no significant release of radioactivity from the tank. The release of all residual primary sodium would result in concentration levels well below the values in 10 CFR 20, Appendix B, Table II for releases to unrestricted areas. Since there is less sodium in the primary sodium storage tanks than in the secondary sodium storage tanks, potential hazard consequences of releasing the contents of a primary sodium tank are bounded by the hypothetical secondary sodium scenario evaluated in the Fermi 1 Safety Analysis Report. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

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, 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, Esquire, Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Branch Chief: Larry W. Camper.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: July 22, 1998, supplemented by October 22, 1998, January 28, May 6 and June 24, 1999.

Description of amendment request: By the referenced submittals the licensee requested the Catawba Technical Specifications be changed to permit the licensee's planned use of fuel supplied by Westinghouse, which has different design characteristics from the fuel currently in use. The staff has previously published two Notices of Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration of Issuance of Amendments. The first notice, dated November 18, 1998 (63 FR 64108), covers the submittals dated July

22 and October 22, 1998. The second notice, dated May 19, 1999 (64 FR 27317), covers the submittal dated May 6, 1999. The June 24, 1999, submittal actually requested an amendment separate from that described above, but nevertheless conveyed a revised proposed Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation'', superseding what was originally proposed in the licensee's previous submittals. Hence, this Notice only covers the revised proposed Figure 2.1.1–1. The Notices referenced above are unaffected.

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 for the June 24, 1999, submittal. The staff has reviewed the licensee's analysis and has performed its own analysis as follows:

First Standard

No. The proposed changes to Figure 2.1.1–1 will not affect the safety function and will not involve any change to the design or operation of any plant system or component. The revised Figure 2.1.1–1 restricts reactor coolant flow to within previously analyzed temperature and pressure conditions. Therefore, no accident probabilities or consequences will be impacted.

Second Standard

No. The proposed changes will not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

Third Standard

No. Margin of safety is associated with confidence in the design and operation of the plant; specifically, the ability of the fission product barriers to perform their design functions during and following an accident. The proposed changes to Figure 2.1.1–1 do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for the proposed change to Figure 2.1.1–1. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: June 24, 1999.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) as follows: (1) Revise Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation,'' which defines the current limits of reactor coolant system (RCS) flow under different combinations of pressure and temperature; (2) revise the Actions associated with Limiting Condition of Operation (LCO) 3.4.1 and Table 3.4.1–1 to reflect the updated assumptions for reactor coolant flow, temperature and pressure; and (3) delete Figure 3.4.1–1, ''RCS Total Flow Rate Versus Rated Thermal Power—Four Loops in Operation,'' since these requirements are being relocated to LOC 3.4.1 and Table 3.4.1–1.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration for the June 24, 1999, submittal, which is presented below:

First Standard

No component modification, system realignment, or change in operating procedure will occur which could affect the probability of any accident or transient. The increase in RCS total flow rate limit will not change the probability of actuation of any Engineered Safety Feature or other device. In order to provide more margin in the core design limits and allow more flexibility for future cycle-specific core design, the analyses that establish these limits were reanalyzed at the proposed TS minimum RCS total flow rate limit. The impact of the power/flow tradeoff is determined for each reanalyzed event either by qualitative evaluation or by explicit reanalysis.

An increase in the Technical Specification minimum RCS total flow rate limit and the revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs [Updated Final Safety Analysis Reports]. The reduced RCS low flow reactor trip setpoint and allowable value will not increase the consequences of the partial loss of forced reactor coolant flow and reactor coolant pump shaft seizure accidents. In these transient reanalyses, the minimum DNBR and peak primary system pressure acceptance criteria are not adversely affected. Therefore, the proposed changes will not involve an increase in the probability or consequences of an accident previously evaluated.

Second Standard

No component modification, system realignment, or change in operating procedure will occur which could create the possibility of a new or different kind of accident. As described in Attachment 3, the proposed increase in Technical Specification minimum RCS total flow rate limit and revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

These amendments will not involve a significant reduction in a margin of safety. As described in Attachment 3, the increase in minimum RCS total flow rate limit and revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs. DNBR, fuel clad intergrity, reactor vessel integrity and containment integrity will not be adversely affected by the proposed changes. Therefore, the proposed changes will not involve any 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: York County Library, 138 East Black Street, Rock Hill, South Carolina.

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

Duke Energy Corporation, et al., Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: July 22, 1998, supplemented by October 22, 1998, January 28, May 6 and June 24, 1999.

Description of amendment request: By the referenced submittals the licensee requested the McGuire Technical Specifications be changed to permit the licensee's planned use of fuel supplied by Westinghouse, which has different design characteristics from the fuel currently in use. The staff has previously published two Notices of

Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration of Issuance of Amendments. The first notice, dated December 16, 1998 (63 FR 69338), covers the submittals dated July 22 and October 22, 1998. The second notice, dated May 19, 1999 (64 FR 35202), covers the submittal dated May 6, 1999. The June 24, 1999, submittal actually requested an amendment separate from that described above, but nevertheless conveyed a revised proposed Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation,'' superseding what was originally proposed in the licensee's previous submittals. Hence this Notice only covers the revised proposed Figure 2.1.1–1. The Notices referenced above are unaffected.

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 for the June 24, 1999, submittal. The staff has reviewed the licensee's analysis, and has performed its own analysis as follows:

First Standard

No. The proposed changes to Figure 2.1.1–1 will not affect the safety function and will not involve any change to the design or operation of any plant system or component. The revised Figure 2.1.1–1 restricts reactor coolant flow to within previously analyzed temperature and pressure conditions. Therefore, no accident probabilities or consequences will be impacted.

Second Standard

No. The proposed changes would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

Third Standard

No. Margin of safety is associated with confidence in the design and operation of the plant; specifically, the ability of the fission product barriers to perform their design functions during and following an accident. The proposed changes to Figure 2.1.1–1 do not involve any change to plant design, operation or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for the proposed change to Figure 2.1.1–1. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: June 24, 1999.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) as follows: (1) Revise Figure 2.1.1–1, ''Reactor Core Safety Limits—Four Loops in Operation,'' which defines the current limits of reactor coolant system (RCS) flow under different combinations of pressure and temperature; (2) revise Table 3.3.1–1 to provide values for the trip setpoint and allowable value for RCS Flow-Low; (3) revise Table 3.3.1–1 to make a typographical correction for T, the nominal T-average at Rated Thermal Power; (4) revise the Actions associated with Limiting Condition of Operation (LCO) 3.4.1 and Table 3.4.1–1 to reflect the updated assumptions for reactor coolant flow, temperature and pressure; and (5) delete Figure 3.4.1–1, ''RCS Total Flow Rate Versus Rated Thermal Power—Four Loops in Operation,'' since these requirements are being relocated to LCO 3.4.1 and Table 3.4.1–1.

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

First Standard

No component modification, system realignment, or change in operating procedure will occur which could affect the probability of any accident or transient. The increase in RCS total flow rate limit will not change the probability of actuation of any Engineered Safety Feature or other device. In order to provide more margin in the core design limits and allow more flexibility for future cycle-specific core design, the analyses that establish these limits were reanalyzed at the proposed TS minimum RCS total flow rate limit. The impact of the power/flow tradeoff is determined for each reanalyzed event either by qualitative evaluation or by explicit reanalysis.

An increase in the Technical Specification minimum RCS total flow rate limit and the revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs [Updated Final Safety Analysis Reports]. The reduced RCS low flow reactor trip setpoint and allowable value will not increase the consequences of the partial loss of forced reactor coolant flow and reactor coolant pump shaft seizure accidents. In these transient reanalyses, the minimum DNBR and peak primary system pressure acceptance criteria are not adversely affected. Therefore, the proposed changes will not involve an increase in the probability or consequences of an accident previously evaluated.

Second Standard

No component modification, system realignment, or change in operating procedure will occur which could create the possibility of a new or different kind of accident. As described in Attachment 3, the proposed increase in Technical Specification minimum RCS total flow rate limit and revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

These amendments will not involve a significant reduction in a margin of safety. As described in Attachment 3, the increase in minimum RCS total flow rate limit and revised power/flow tradeoff will not adversely affect the steady-state or transient analyses documented in Chapters 3, 4, 6, and 15 of the McGuire and Catawba Nuclear Station UFSARs. DNBR, fuel clad integrity, reactor vessel integrity and containment integrity will not be adversely affected by the proposed changes. Therefore, the proposed changes will not involve any 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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

Entergy Operations, Inc. (EOI), Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: July 14, 1999.

Description of amendment request: The proposed amendments delete requirements from the Technical Specifications to maintain a Post Accident Sampling System (PASS). Licensees were required to implement PASS upgrades as a result of NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades were an outcome of the NRC's lessons learned from the accident that occurred at Three Mile Island, Unit 2. EOI has stated that the information obtained using PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

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

Criterion 1—[The Proposed Change] Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the Three Mile Island Unit 2 (TMI–2) accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that the actual benefits afforded by a PASS provide little benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides no benefit to the plant staff in coping with an accident. The use of the PASS may be counter productive to plant operations since its operation will divert resources away from accident management, the sample results may be ambiguous and may be misinterpreted, and the use of PASS may restrict personnel movements in certain areas of the plant while resulting in additional fission product release points outside the containment.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Additionally, preliminary discussions with the State of Arkansas have indicated that the elimination of the PASS will not adversely impact actions taken by the State during an emergency event. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 [accident] guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan PARs [protective action recommendations].

Therefore, the elimination of PASS requirements of the ANO–1 and ANO–2 [Arkansas Nuclear One, Unit 1 and Unit 2] Technical Specifications (TS) and subsequent requested relief from the requirements of NUREG–0737 and Regulatory Guide 1.97, Revision 3, does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2—[The Proposed Change] Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The relief from PASS related NUREG–0737 and Regulatory Guide 1.97 requirements in addition to the proposed TS changes will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—[The Proposed Change] Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety at ANO–1 and ANO–2. Non-PASS methodologies are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events nor rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations, Inc. has determined that the requested change 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: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

Indiana Michigan Power Company, Docket Nos. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan.

Date of amendment requests: December 3, 1998.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3/4.7.7, ''Sealed Source Contamination,'' and the associated bases to address testing requirements for fission detectors. The proposed changes would provide consistency between the unit 1 and Unit 2 TS requirements and with NUREG–0452, ''Standard Technical Specifications.''.

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

Criterion 1

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

The proposed changes clarify testing requirements for fission detectors. When the fission detectors are tested for surface contamination, they do not interfere with plant equipment and they do not affect plant operation. The detectors are not assumed to initiate an accident; therefore, the probability of an accident previously evaluated is not changed.

Conducting tests prior to using a new fission detector provides assurance that intake limits will not be exceeded. There is no change to the nuclear material contained in the detector. The fission detectors are not used to mitigate the consequences of postulated accidents. Therefore, the consequences of an accident remain the same as previously evaluated.

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

Criterion 2

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

The proposed changes do not affect the design or operation of systems, structures, or components in the plant. There are no changes to parameters governing plant operation, and no new or different types of equipment will be installed. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

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

The proposed changes do not introduce new equipment, equipment modifications, or new or different modes of plant operation. These changes do not affect the operational characteristics of any equipment or systems.

Therefore, it is concluded that these changes do not involve a significant reduction in the margin of safety.

Conclusion

In summary, based upon the above evaluation, the Licensee has concluded that these changes involve no significant hazards consideration.

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

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

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

NRC Section Chief: Claudia M. Craig.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: March 29, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification Surveillance Requirement 3.9.1.1 and the associated Bases 3.9.1 to delete the requirement for the refuel platform fuel grapple fully retracted position interlock.

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 change removes a redundant interlock and will not impact the functionality of associated interlocks. The removal of the ''refuel platform fuel grapple fully retracted position'' refueling interlock will not affect the ability of the remaining refueling interlocks to produce a rod block during fuel moves. The administrative controls in place do not allow control rod withdrawals while fuel is being moved or fuel movement while rods are withdrawn. The fuel grapple full up interlock is a redundant and diverse interlock and its removal has no impact on plant safety. The interlock's intent, to provide a backup to the load sensor, is not required since the setpoint is currently low enough to provide adequate protection therefore not significantly increasing the probability of an accident previously evaluated.

The refueling interlocks are not used to prevent or to mitigate the fuel handling accident as discussed in the PBAPS [Peach Bottom Atomic Power Station], Units 2 and 3, UFSAR [Updated Final Safety Analysis Report], Section 14.6.4 (''Refueling Accident''). The ''refuel platform fuel grapple fully retracted position'' interlock and the ''refuel platform fuel grapple, fuel loaded'' interlock both provide rod blocks during fuel movement over the core. Additionally, the refueling interlocks are not assumed as an initial condition in the control rod drop accident as discussed in the PBAPS, UFSAR, Section 14.6.2 (''Control Rod Drop Accident''). The control rod drop accident is only analyzed when the reactor is critical and not during refueling operations.

The refueling interlocks associated with the refueling platform provide rod blocks to ensure that control rods can not be withdrawn when fuel is being moved over the core (PBAPS, Units 2 and 3, UFSAR Section 14.5.3.3, ''Control Rod Removal Error During Refueling''). They are also used to prevent refueling bridge motion towards the core if a control rod is withdrawn during fuel movements (PBAPS, Units 2 and 3, UFSAR Section 14.5.3.4, ''Fuel Assembly Insertion Error During Refueling''). These interlocks prevent the possibility of an inadvertent criticality during refueling. However, removal of the ''refuel platform fuel grapple fully retracted position'' interlock, which is a redundant and diverse interlock, will not prevent the remaining interlocks from performing their intended safety functions. The refueling interlocks are active with the mode switch in refuel, and are only designed to reinforce administrative procedures for moving fuel. Therefore, the proposed TS changes will not involve a significant increase in the probability of an accident previously evaluated.

The fuel or core loading characteristics are not altered by the removal of this interlock. The dose resulting from a potential control rod withdrawal or fuel bundle error event is not increased as a result of eliminating this redundant and diverse interlock. Therefore, the removal of the ''refuel platform fuel grapple fully retracted position'' interlock will not involve a significant increase in the probability or 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 refueling interlocks are not accident initiators. Nor will any new failure mode be introduced by the removal of the ''refuel platform fuel grapple fully retracted position'' interlock. The interlocks are used to reinforce administrative controls which prevent fuel movement over the core with control rods withdrawn and preclude withdrawal of control rods when the fuel is being moved over the core. The interlock for ensuring the fuel grapple is fully up, is a redundant and diverse interlock since a load sensor determines if the main hoist is loaded with a fuel bundle. This redundant and diverse interlock prevents the withdrawal of a control rod while moving fuel during refueling. The setpoint is low enough to ensure a rod block will be received if the main hoist is being used to move fuel over the core and to prevent movement of the refueling bridge. The remaining refueling interlocks, in combination with the refueling procedures, will still prevent an inadvertent criticality during refueling operations. Fuel handling procedures require that interlocks be verified by observing the rod withdraw permissive light in the control room, and by monitoring the rod block interlock light on the refuel bridge. Therefore, the proposed TS changes do 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.

This change will not involve a significant reduction in a margin of safety. The ''refuel platform fuel grapple fully retracted position'' interlock is redundant and diverse to the ''refuel platform fuel grapple, fuel loaded'' interlock on the main hoist. The other two hoists on the bridge have the fuel loaded interlock but do not have the backup full up position interlock. The margin of safety of the refueling interlocks will not be significantly reduced by this change since redundant interlocks are not required (this a nonsafety-related function) and the original justification for using it, a high load weight setpoint, is no longer applicable. The system consists of a single channel, and no current design basis for using redundant and diverse interlocks to provide the rod block. Additionally, the Reactor Manual Control System will not be affected by this change. The system's ability to provide a rod block is not affected by this change.

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

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

Attorney for Licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: April 5, 1999

Description of amendment request: The proposed changes would revise Appendix A (Section 6.1) and Appendix B (Section 7.1) of the James A. FitzPatrick Technical Specifications. The proposed changes would remove the position title of General Manager from these sections and would state that if the Site Executive Officer (SEO) is unavailable, he will delegate his responsibilities to another staff member, in writing. In addition the position title of Resident Manager, used in Apendix B, Section 7.1, would be replaced by Site Executive Officer.

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:

Consistent with the criteria of 10 CFR 50.92, the proposed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: The proposed changes to Appendix A (Section 6.1) and Appendix B (Section 7.1) are administrative in nature in that they do not change the intent of the Technical Specifications. If the SEO is unavailable, he will still delegate his responsibilities to a qualified personnel member, such as the Plant Manager or one of the General Managers. These changes can not cause an accident or contribute to the probability or consequences of one.

The replacement of the position title of Resident Manager with Site Executive Officer in Appendix B, Section 7.1, was already approved by the NRC in Amendment 228.

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

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: The proposed changes to Appendix A (Section 6.1) and Appendix B (Section 7.1) are administrative in nature as they do not affect the function of plant equipment or the way the equipment operates. The changes do not change the intent of the current TS, in that if the SEO is unavailable, he will delegate his responsibilities to another personnel member such as the Plant Manager or one of the General Managers. Appendix A (Section 6.1) and Appendix B (Section 7.1) are being revised to eliminate the need for future TS changes to these sections resulting solely from the creation of new or revised management positions (such as the Plant Manager), title changes to the position of General Manager, or a change to the number of General Managers. These types of organizational changes will be evaluated using the criteria of 10 CFR 50.59.

The replacement of the position title of Resident Manager with Site Executive Officer in Appendix B, Section 7.1, was already approved by the NRC in Amendment 228.

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

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: The proposed changes to Appendix A (Section 6.1) and Appendix B (Section 7.1) are administrative changes associated with the delegation of the SEO's responsibilities when he is unavailable. These changes do not change the intent of the current TS, in that in the SEO's absence, he will still delegate his responsibilities to other personnel members such as the Plant Manager or General Managers.

The replacement of the position title of Resident Manager with Site Executive Officer in Appendix B, Section 7.1, was already approved by the NRC in Amendment 228.

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

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

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: June 22, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications by changes to the Pressure and Temperature (P–T) limits. As part of this proposed change the licensee is proposing to add separate bottom head curves A_{BH} and B_{BH} for in-service hydrostatic and leak tests and non-nuclear heatup and cooldown, respectively. In addition, a non-beltline curve (i.e., A_{NB}) for in-service hydrostatic and leak tests is being proposed.

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

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

The changes to the P–T curves are being proposed to preclude brittle fracture of RPV [Reactor Pressure Vessel] materials for up to 32 EFPY [effective full-power years]. In addition to the P–T curve for up to 32 EFPY, a P–T curve has been prepared for exposures up to 24 EFPY to shorten outage time for startups conducted prior to reaching this exposure. Safety margins specified in 10 CFR 50, Appendix G and Appendix G to Section XI of the ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] will continue to be met for each of these curves. Therefore, there is not a significant increase in the probability of an accident previously evaluated.

The RPV, as part of the reactor coolant system, provides a barrier to the release of reactor coolant. Operation in accordance with the proposed amendment will preclude brittle fracture of the RPV consistent with current requirements, and consequently, does not significantly increase the consequences of an accident previously evaluated.

Based on the above, operation of the FitzPatrick plant in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alterations to plant configurations or introduce any new accident precursors which could initiate a new or different kind of accident. The proposed change does not affect the intended function of the RPV nor does it affect the operation of the RPV in a way which would create a new or different kind of accident. The changes to the P–T curves are being proposed to preclude brittle fracture of RPV materials for up to 32 EFPY. Safety margins specified in 10 CFR 50, Appendix G and Appendix G to Section XI of the ASME Code will continue to be met. Therefore, operation of the FitzPatrick plant in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The existing FitzPatrick P–T curves were developed using safety margins for brittle fracture found in 10 CFR 50 Appendix G. The proposed FitzPatrick P–T curves, which are valid for up to 32 EFPY of operation, were also developed using safety margins for brittle fracture found in 10 CFR 50 Appendix G. Based on this, operation of the FitzPatrick plant in accordance with the proposed amendment will continue to preclude brittle fracture of the RPV materials during in-service hydrostatic and leak tests, non-nuclear heatup and cooldown, and core critical operation without a significant reduction in a margin of safety. Therefore, operation of the FitzPatrick plant in accordance with 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 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

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: July 2, 1999.

Description of amendment request: The proposed amendments would relocate the requirements from Technical Specification 3/4.3.4, ''Instrumentation, Turbine Overspeed Protection,'' and the associated bases to licensee-controlled documents in accordance with Generic Letter 95–10, ''Relocation of Selected Technical Specifications Requirements Related to Instrumentation.''

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

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

The requested amendments will not involve an increase in the probability or consequences of an accident previously evaluated. Relocation of the affected Technical Specification sections and their Bases to the Salem UFSAR [Updated Final Safety Analysis Report] will have no affect on the probability that any accident will occur. Additionally, the consequences of an accident will not be affected because the Turbine Overspeed Protection system will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

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

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of the relocation of the Turbine Overspeed Protection system Technical Specification requirements and their Bases to the Salem UFSAR. Plant operation will not be affect by the proposed amendments and no new failure modes will be created.

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

The proposed amendments will not involve a reduction in the margin of safety. Relocation of the affected Technical Specification requirements to the Salem UFSAR is consistent with NUREG 1431, Standard Technical Specifications—Westinghouse Plants which do not include Technical Specification requirements for the Turbine Overspeed Protection system. The proposed amendments are consistent with the NRC philosophy of encouraging utilities to propose amendments that are consistent with NUREG 1431.

Based on the above, the proposed changes 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 Section Chief: James W. Clifford.

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: July 16, 1999.

Description of amendment request: Proposed Technical Specifications (TS) change to increase the action requirement time to be in Mode 3 if the temperature of the ultimate heat sink (UHS) exceeds the TS limit of 75 °F. The increased time will only apply if the UHS temperature is between 75 and 77 °F. The Bases for the associated TS will also be revised.

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

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

The proposed changes will allow plant operation to continue for an additional 12 hours with the temperature of the Ultimate Heat Sink (UHS) up to 2 °F above the Technical Specification limit of 75 °F. This increase in UHS temperature will not affect the normal operation of the plant to the extent which would make any accident more likely to occur. In addition, there exists adequate margin in the safety systems and heat exchangers to assure the safety functions are met at the higher temperature. An evaluation has confirmed that safe shutdown will be achieved and maintained for a loss of coolant accident (LOCA) with a loss of normal power (LNP) and a single active failure with a UHS water temperature as high as 77 °F.

The proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed changes can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will allow plant operation to continue for an additional 12 hours with the temperature of the UHS up to 2 °F above the Technical Specification limit of 75 °F. This will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is unaffected by the changes. In addition, the UHS is not an accident initiator. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes will allow plant operation to continue for an additional 12 hours with the temperature of the UHS up to 2 °F above the Technical Specification limit of 75 °F. Evaluations have been performed which demonstrate that the safety systems have adequate margin to ensure their safety functions can be met with a UHS temperature of 77 °F. In addition, safe shutdown capability has been demonstrated for a UHS water temperature as high as 77 °F.

The proposed changes will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, 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 Section Chief: James W. Clifford.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: July 12, 1999.

Description of amendment request: The proposed change will revise Technical Specifications (TSs) TS 2.1.1.2, ''Reactor Core [Safety Limits] SLs,'' and Section 5.6.5, ''Core Operating Limits Report.'' These Sections will be revised to: (1) Incorporate revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the use of a cycle-specific analysis performed by General Electric Nuclear Energy (GENE) for Peach Bottom Atomic Power Station, Unit 3, (PBAPS, Unit 3) Cycle 13, (2) delete previously added footnotes which are no longer necessary, and (3) update a reference contained in TS 5.6.5.b.2 which documents an analytical method used to determine the core operating 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 TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the cycle specific SLMCPRs for incorporation into the TS, and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999 safety evaluation report. This change in SLMCPRs cannot increase the probability or severity of an accident.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The fuel licensing acceptance criteria for the SLMCPR calculation apply to PBAPS, Unit 3, Cycle 13 in the same manner as they have applied previously. The probability of fuel damage is not increased. Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

In addition to the change to the SLMCPR, the footnotes to TS 2.1.1.2 and TS 5.6.5.b.1 are being deleted. The footnote associated with TS 2.1.1.2 was originally included to ensure that the SLMCPR value was only applicable for the identified cycle. The footnote was added to TS 5.6.5.b.1 because Amendment 25 and the R-factor calculation methodology were not yet NRC approved. Amendment 25 and the R-factor methodology have subsequently been approved. Therefore, these footnotes are no longer necessary. The footnotes were for information only, and have no impact on the design or operation of the plant. The deletion of the footnotes associated with TS 2.1.1.2 and TS 5.6.5.b.1 is an administrative change that does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Revision 1 ARTS/MELLLA [Maximum Extended Load Line Limit and ARTS Improvement Program Analysis for Peach Bottom Atomic Power Station Unit 2 and 3,] analysis contained in TS 5.6.5.b.2 is being updated to a Revision 2 analysis, to reflect changes that were previously approved by the NRC as documented in the safety evaluation report dated August 10, 1994 (Amendment No. 192 for PBAPS, Unit 2). This is an administrative change which will ensure that the references contained in the PBAPS Technical Specifications are accurate and consistent with other licensing documents. No technical changes are occurring which have not been previously approved by the NRC. Therefore, this change does not involve a significant increase in the probability or 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 SLMCPR is a TS numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core during the limiting postulated accident. The new SLMCPRs are calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Additionally, this proposed change will delete footnotes contained in TS 2.1.1.2 and TS 5.6.5.b.1 as the result of the NRC approval of analysis associated with Amendment 25 and the R-factor methodology. The proposed change also updates the ARTS/MELLLA analysis contained in TS 5.6.5.b.2. This revision contains information which was previously approved by the NRC. Therefore, the deletion of the footnotes associated with TS 2.1.1.2 and TS 5.6.5.b.1, and the updating of the reference contained in TS 5.6.5.b.2 are administrative changes that do not create the possibility of a new or different kind of accident from any previously evaluated.

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

There is no significant reduction in the margin of safety previously approved by the NRC as a result of: (1) the proposed changes to the SLMCPRs, (2) the proposed change that will delete the footnotes to TS 2.1.1.2 and TS 5.6.5.b.1, and (3) updating the reference to the ARTS/MELLLA analysis contained in TS 5.6.5.b.2. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. The fuel licensing acceptance criteria for the calculation of the SLMCPR apply to PBAPS, Unit 3 Cycle 13 in the same manner as they have applied previously. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes will not involve a significant reduction in the margin of safety previously approved by the NRC.

Additionally, the proposed changes that delete the footnotes to TS 2.1.1.2 and TS 5.6.5.b.1, and update the revision to the ARTS/MELLLA analysis contained in TS 5.6.5.b.2, are administrative changes that will not significantly reduce the margin of safety previously approved by the NRC.

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

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

Attorney for Licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 28, 1999.

Description of amendment request: The proposed amendment would revise the Improved Technical Specifications (ITS) associated with the Reactor Coolant System (RCS) Leakage Detection Instrumentation.

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

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes add further requirements for redundancy and a requirement to perform either an RCS water inventory balance or analyses of containment atmosphere grab samples once within 12 hours and every 12 hours thereafter when the particulate containment atmosphere radioactivity monitor is unavailable while in Modes 1, 2, 3, and 4. This does not increase the probability of an accident previously evaluated since the compensatory actions are either a calculation utilizing installed indication or the measurement of a sample drawn downstream from the containment atmosphere sample isolation valves and are of themselves not an accident initiator. The proposed compensatory actions are based on the NUREG–1431 guidance and the proposed frequencies are more conservative, which gives a higher assurance that the RCS leakage rate can be adequately monitored.

Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes add further requirements for redundancy and the proposed change for compensatory actions when the particulate containment atmosphere radioactivity monitor is inoperable does not of itself involve a physical alteration of the plant (ie. no new or different type of equipment will be added to perform the required actions) or changes in the methods governing normal plant operation. The changes only involve implementing currently approved alternate methods to determine the RCS leak rate on an increased frequency. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only add conservatism in the number of required RCS leakage detection instrumentation and add more conservative compensatory actions that are to be taken when the containment atmosphere particulate radioactivity monitor is inoperable. The compensatory actions are based on the guidance of NUREG–1431. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in 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. Therefore, it is concluded that the proposed changes meets the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

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

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

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment requests: September 10, 1998 (PCN–496), as supplemented July 19, 1999.

Description of amendment requests: The proposed amendments would modify the Technical Specifications for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 to delete the requirements for equipment used to control hydrogen in the containment structure.

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

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

Response: No

The containment hydrogen control system is currently classified as an engineered safety feature that serves as the combustible gas control system in the containment. The hydrogen control system is composed of a hydrogen recombiner subsystem and a hydrogen purge subsystem. Hydrogen control subsystem components are not considered to be accident initiators.

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

The hydrogen control system is provided to ensure that the hydrogen concentration is maintained below the flammability limit of 4% so that containment integrity is not challenged following a design basis Loss Of Coolant Accident (LOCA). Existing analysis show[s] that the hydrogen concentration will not reach the flammability limit of 4% for at least 13.5 days after a design basis LOCA. The time available will be extended to over 30 days using more realistic hydrogen generation rates. The containment peak pressure will remain below the San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 & 3) containment design pressure of 60 psig [pounds per square inch gauge] during this time. Beyond 30 days, hydrogen concentration may reach the flammability limit. However, containment failure due to hydrogen combustion is unlikely based on the results of the SONGS 2 & 3 IPE [indvidual plant examination] study. The detailed

SONGS 2 & 3-specific containment integrity analysis indicates that containment rupture pressure is approximately 139 psig with 95% confidence. Therefore, this change does not increase the consequences of accidents previously evaluated.

Removal of the existing requirements for hydrogen control will eliminate the Emergency Operating Instruction (EOI) steps for hydrogen control and hence simplify the EOIs. This would have a positive impact on public health risk by reducing the probability of operator error during potential accidents and hence reduce the core damage frequency. As proposed in this change request, these changes will allow the operators to address all hydrogen control issues as part of the proposed Accident Management Guidelines which cover operator actions at long time frames following accidents.

Removal of the existing requirements for hydrogen control will eliminate the EOI steps to initiate the containment hydrogen purge. This will result in a lower probability of a failed open containment purge valve. Consequently, the offsite doses would be reduced due to the reduction of the probability of a failed-open containment purge valve. The changes described in this request result in a ''risk positive'' change.

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

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

Response: No

This proposed change does not change the design or configuration of the plant beyond the hydrogen control system. Hydrogen generation following a design basis LOCA has been evaluated in accordance with regulatory requirements. Deletion of the hydrogen control system from the Technical Specifications does not alter the hydrogen generation processes post-LOCA. The consideration of hydrogen generation will no longer be included in the design basis of SONGS 2 & 3. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

The changes described in this change request result in a ''risk positive'' change. Removal of the existing requirement for a hydrogen control system will, by eliminating the EOI steps for hydrogen control, result in lower operator error probabilities. Elimination of the EOI steps to initiate the containment hydrogen purge will result in a lower probability of a failed-open containment purge valve, resulting in lower large early release probabilities.

Therefore, this change involves an increase in safety, not a reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: April 13, 1999.

Description of amendment request: Southern Nuclear Operating Company (SNC) proposes to revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS) Limiting Condition for Operation (LCO) Applicability LCO 3.0.4 and Surveillance Requirement (SR) Applicability SR 3.0.4. The proposed changes would update the versions of LCO 3.0.4 and SR 3.0.4 that appear in the existing VEGP TS to be consistent with the versions of LCO 3.0.4 and SR 3.0.4 as they appear in Revision 1 to NUREG–1431. The proposed change would add the words ''or that are part of a shutdown of the unit,'' to LCO 3.0.4 to allow reactor shutdowns that are not necessarily required by other TS Required Actions.

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

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

No. The proposed change has impact on what equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to unit shutdowns or entry into MODES 5 and 6. This change could increase the probability or consequences of an accident previously evaluated if applied without consideration to all applicable transitions. However, as part of the change, an evaluation is attached in the form of a matrix that identifies those specifications to which LCO 3.0.4 and SR 3.0.4 must continue to apply. Therefore, only those specifications that do not impact safety for these plant conditions are afforded this relaxation. As such, there is no increase in the probability or consequences of an accident previously evaluated as this assessment has been performed and documented with the submittal.

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

No. The proposed change administratively changes when equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to unit shutdown or entry into MODES 5 and 6. However, as no changes in equipment function or operation are included, there is no increase in the probability of a new or different kind of accident from those previously evaluated.

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

No. The proposed change has impact on what equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to unit shutdown or entry into MODES 5 and 6. This change could impact the margin of safety of some accidents if applied without consideration to all applicable transitions. However, as part of the change, an evaluation is attached in the form of a matrix, that identifies those specifications to which LCO 3.0.4 and SR 3.0.4 must continue to apply. Therefore, only those specifications that do not impact safety for these plant conditions, which includes any impact on margin of safety are afforded this relaxation. As such, there is 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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia.

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: April 28, 1999.

Description of amendment request: The amendments revise Vogtle's licensing basis to allow the licensee to establish containment hydrogen monitoring within 90 minutes of initiation of a safety injection following a loss-of-coolant accident, compared to the current 30 minutes requirement.

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

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

Containment hydrogen concentration is not an input parameter to the FSAR Chapter 15 accident analyses for a loss of reactor or secondary coolant accidents; nor is it used as an initial assumption for the containment response analysis. Control room operators use the containment hydrogen monitors to establish hydrogen control measures should it become necessary. However, the actions required to establish containment hydrogen monitoring are a distraction for the operators from more important tasks during the early phases of an accident. Hydrogen production occurs over a long period and a significant accumulation is not expected for several hours into the event. This function is more appropriately included as a part of the long-term core damage assessment process. The one-hour extension will have a positive impact on the ability of the operators to concentrate on their more immediate actions while having no negative impact on the long-term assessment efforts. Therefore, the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Operation of the containment hydrogen monitors is not an initiator of any design basis accident. Control room operators use the containment hydrogen monitors following a LOCA to establish hydrogen control measures should it become necessary. Accurate indication of containment hydrogen concentration is needed prior to initiating recombiner operation or containment venting and for long-term core damage assessment. The proposed license amendment would not eliminate the requirement to establish hydrogen monitoring, but would permit it to be delayed until those actions required to diagnose the event and verify proper operation of essential safety equipment have been completed. The one-hour extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment will not create a new or different kind of accident from any previously evaluated.

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

The need to establish hydrogen control measures will not be present within the first 90 minutes following a LOCA since there will not be significant hydrogen accumulation. By extending the time allowed to establish containment hydrogen monitoring, the operators can remain focused on the actions necessary to assess and mitigate the accident before redirecting their attention to long-term recovery actions. The one-hour extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment will not involve a significant reduction in a margin of safety, but will instead result in an overall enhancement to 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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia.

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: May 18, 1999.

Description of amendment request: The proposed change would revise Surveillance Requirements (SRs) 3.8.1.3 and 3.8.1.13 to reduce the loading requirements for the diesel generators (DGs). Presently, SR 3.8.1.3 requires that the DGs be loaded and operated for greater than or equal to 60 minutes between 6800 kW and 7000 kW at least once every 31 days. The proposed change would revise the lower end of the load band in SR 3.8.1.3 to 6500 kW from 6800 kW. Revised SR 3.8.1.3 would require that the DGs be loaded and operated for greater than or equal to 60 minutes at a load greater than or equal to 6500 kW and less than or equal to 7000 kW at least once every 31 days.

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

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

No. The proposed change affects only the DG loading requirements (kW and kVAR) specified in SRs 3.8.1.3 and 3.8.1.13. These loading requirements have no impact on or relationship to the probability of any of the initiating events assumed for the accidents previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated. Furthermore, since the proposed loading requirements bound the maximum expected loading for the DGs, SRs 3.8.1.3 and 3.8.1.13 will continue to demonstrate that the DGs are capable of performing their safety function. Since the proposed change does not adversely affect the capability of the DGs to perform their safety function, the outcomes of the accidents previously evaluated (i.e., radiological consequences) will not be affected. Therefore, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

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

No. The proposed change affects only the DG loading requirements (kW and kVAR) specified in SRs 3.8.1.3 and 3.8.1.13. The proposed change will not introduce any new equipment or create new failure modes for existing equipment. Other than the reduced loading requirements for the DGs, the proposed change will not affect or otherwise alter plant operation. The DGs will remain capable of performing their safety function. No other safety related or important to safety equipment will be affected by the proposed change. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed change reduces the loading requirements of SRs 3.8.1.3 and 3.8.1.13. With one exception, the new loading requirements are consistent with the latest regulatory guidance found in Regulatory Guide (RG) 1.9, Revision 3, ''Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants,'' July 1993. The one exception to RG 1.9, the loading requirements for the 2-hour portion of the endurance and margin test (SR 3.8.1.13), will require testing at loads in excess of 105 percent of the maximum expected load as opposed to 105 percent of the continuous duty rating. Testing for at least 2 hours at 105 percent of the maximum expected load will continue to demonstrate adequate margin, and it will reduce wear and tear on the DGs due to testing. Reduction in wear and tear should inherently increase the reliability of the DGs. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: June 24, 1999 (TS 99–05)

Brief description of amendments: The proposed amendments would change the Sequoyah Units 1 and 2 Technical Specification (TS) requirements by clarifying and changing the surveillance requirements for the ice weight in the ice condenser baskets. This request is a lead-plant change for all Westinghouse-designed ice condenser plants and will be incorporated into the Improved Standard Technical Specifications (ISTSs), if approved.

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

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

The proposed TS amendments discussed below cannot increase the probability of occurrence of any analyzed accident because they are not the result or cause of any physical modification to the ice condenser structures, and for the current design of the ice condenser, there is no correlation between any credible failure and the initiation of any previously analyzed accident.

Regarding the consequences of analyzed accidents, the proposed amendment provides for consistency with the ISTSs by: (1) requiring the actions if one or more ice condenser ice baskets are determined to weigh below the minimum specified value to be made a part of the TS surveillance requirement (SR) instead of being located in the bases, and (2) relocating the ice basket selection methodology into the bases. This ensures consistent interpretation of the requirements of the TS in accordance with the ISTSs. The clarification of the response required if one or more ice baskets in a given bay are determined to be underweight ensures sufficient ice is maintained in each bay to prevent early meltout in a local zone following a design basis accident (DBA) and that the required overall ice weight is maintained in the ice condenser. The relocation of the ice basket selection methodology to the bases does not result in any change to the intent or implementation of this portion of the TSs since plant procedures ensure the requirements of the bases of the TSs are correctly implemented. Additionally, the clarification that the weight requirement is applicable to the beginning of the cycle does not change the present intent of the TS, but ensures there is no confusion, since the weight at the end of the operating cycle may be less than that specified in the SR due to sublimation. This does not result in a change to the intent or implementation of the TS since a sublimation allowance was provided in the original SR weight requirement. These clarifications do not result in any [effect] on plant equipment or operation and the actions taken during the implementation of the revised TS will be the same as prior to the revision. Therefore, the clarification of these requirements will not increase the consequences of any accident previously evaluated.

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

The inclusion of the action required for an underweight ice basket in the TS SR, instead of in the bases of the TS, provides for the consistent interpretation of the requirement. The clarification of the response required if one or more ice baskets in a given bay are determined to be underweight ensures sufficient ice is maintained in each bay to prevent early meltout in a local zone following a DBA and that the required overall ice weight is maintained in the ice condenser. The relocation of the ice basket selection methodology to the bases does not result in any change to the intent or implementation of this portion of the TSs since plant procedures ensure the requirements of the bases of the TSs are correctly implemented. Additionally, the clarification that the weight requirement is applicable to the beginning of the cycle does not change the present intent of the TS, but ensures there is no confusion, since the weight at the end of the operating cycle may be less than that specified in the SR due to sublimation. This does not result in a change to the intent or implementation of the TS since a sublimation allowance was provided in the original SR weight requirement. The operation, design and maintenance of the ice condenser and its associated equipment will not change as a result of these clarifications. Therefore, the implementation of these clarifications will not create the possibility of accidents or equipment malfunctions of a new or different kind from any previously evaluated.

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

The proposed amendment allows for the consistent interpretation of the required actions if an ice basket is determined to weigh less than the required minimum. The inclusion of these actions in the TS SR instead of in the TS bases assures the correct actions will be taken as intended by the TSs. The clarification of the response required if one or more ice baskets in a given bay are determined to be underweight ensures sufficient ice is maintained in each bay to prevent early meltout in a local zone following a DBA and that the required overall ice weight is maintained in the ice condenser. The relocation of the ice basket selection methodology to the bases does not result in any change to the intent or implementation of this portion of the TSs since plant procedures ensure the requirements of the bases of the TSs are correctly implemented. Additionally, the clarification that the weight requirement is applicable to the beginning of the cycle does not change the present intent of the TS, but ensures there is no confusion, since the weight at the end of the operating cycle may be less than that specified in the SR due to sublimation. This does not result in a change to the intent or implementation of the TS since a sublimation allowance was provided in the original SR weight requirement. The proposed clarifications do not result in or have any [effect] on the operation, design, or maintenance of any plant equipment. Thus the design limits for the continued safe function of the containment structure following a DBA are not exceeded due to this change; therefore, the proposed amendment does not involve a reduction in a margin of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: June 25, 1999 (TS 99–004).

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant Unit 1 Technical Specifications (TS) and associated TS Bases for Limiting Condition for Operation (LCO) 3.7.1, Main Steam Safety Valves, to provide a new requirement to reduce the Power Range Neutron Flux-High reactor trip setpoints when two or more main steam safety valves (MSSVs) per steam generator are inoperable. This proposal is based on a generic change developed by the Westinghouse Owners Group (WOG), TSTF–235, Revision 1, which has been approved by the NRC staff.

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

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

The proposed change to TS LCO 3.7.1 requires a reduction of the Power Range Neutron Flux-High reactor trip setpoints to a corresponding power level depending on the number of inoperable MSSVs. The change is based on and consistent with an industry sponsored change (TSTF–235, Revision 1) which has been reviewed and accepted by the NRC staff.

Although plant procedures currently require resetting the high flux trip, it is not a TS requirement. The proposed amendment will provide a more appropriate barrier to prevent the plant from being operated under a non-conservative technical specification action statement in a region where multiple inoperable MSSVs coincident with a reactivity insertion event such as an inadvertent rod cluster control assembly (RCCA) bank withdrawal could result in overpressurization of the secondary system.

No change is made in the probability of initiating accident, i.e., RCCA bank withdrawal, and by requiring the reactor trip setpoint reduction, a potential mismatch between core power and turbine load without sufficient steam relief capacity is eliminated. Therefore, the change requested by this amendment actually decreases the consequences of an accident previously evaluated (without credit for procedure actions to reduce the trip setpoints).

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

Without crediting existing plant procedures, the addition of the proposed TS change prevents the plant from being operated in a region where an overpressurization of the main steam system is postulated to potentially occur. The proposed change assures that the existing FSAR [Final Safety Analysis Evaluation Report] accident analysis remains bounding for events that challenge the relieving capacity of the MSSVs. Since the addition of the TS action adds a more appropriate administrative barrier to prevent operation in an undesired region and because the change is bounded by the current accident analysis described in the FSAR, a new or different kind of accident has not been created as a result of this license amendment.

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

The proposed TS change eliminates a non-conservative TS action to prevent the plant from being operated in a region where an overpressurization of the main steam system is postulated to potentially occur. Since the addition of the TS action adds a more effective administrative barrier to prevent operation in an undesired region and because the change is bounded by the existing FSAR accident analysis, the margin of safety has actually increased for the proposed change. For these reasons, the proposed 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 amendment request involves no significant hazards consideration.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 8, 1999.

Description of amendment request: The amendment request proposes to increase the allowable values for the engineered safety features actuation system (ESFAS) loss-of-power 4 kV undervoltage trips in the current Technical Specifications (TSs) Table 3.3–4 (functional units 8.a and 8.b) and in Surveillance Requirement (SR) 3.3.5.3 of the improved TSs. The word ''nominal'' is also being added to describe the trip setpoint in SR 3.3.5.3 and in the Bases of the improved TSs. The improved TSs were issued in Amendment 123 dated March 31, 1999, but have not yet been implemented.

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

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

The reactor protection system performance will remain within the bounds of the previously performed accident analysis. 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 is no change to the technical specification trip setpoints; therefore, there is 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 and be no change to normal plant operating parameters or accident mitigation capabilities. The allowable values and the trip setpoints in the protection system proposed to be changed are not initiators of accidents previously evaluated.

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

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

There are no changes in the method by which any safety related plant system performs its safety function. The normal manner of plant operation remains unchanged because the methodology to determine the allowable value and the trip setpoints remains unchanged. The increase in allowable value for the trip setpoints still provides margin between the nominal trip setpoint and allowable value while taking into account worst case 4.16 kV Class 1E system (NB) bus voltages that could be possible during steady state loss-of-coolant accident (LOCA) conditions. The change in allowable value for the undervoltage protection functions does not impact the systems capability to:

a. Trip the 4.16 kV preferred normal and alternate bus feeder breakers to remove the deficient power source to protect the Class 1E equipment from damage;

b. Shed all loads from the bus except the Class 1E 480 Vac load centers and centrifugal charging pumps to prepare the buses for re-energization by the load shedder and emergency load sequencer (LSELS); and

c. Generate a emergency diesel generator (EDG) start signal.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The allowable values and the trip setpoints in the protection system proposed to be changed are not initiators of accidents. 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 undervoltage protection functions are to:

a. Trip the 4.16 kV preferred normal and alternate bus feeder breakers to remove the deficient power source to protect the Class 1E equipment from damage;

b. Shed all loads from the bus except the Class 1E 480 Vac load centers and centrifugal charging pumps to prepare the buses for re-energization by the load shedder and emergency load sequencer (LSELS); and

c. Generate a EDG start signal.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change in the safety analysis limit. There will be no effect on the manner in which safety limits or engineered safety features actuation system settings are determined nor will there be any affect on those plant systems necessary to assure the accomplishment of the above protection functions. Therefore, there will not be a significant reduction in a margin of safety.

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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

Previously Published Notice of Consideration of Issuance of Amendment to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity for a Hearing

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

Date of amendment request: June 30, 1999.

Brief description of amendment: The proposed amendment would revise Technical Specification (TS) 3/4.7.5 of the current TSs by adding a temporary action statement that would allow the plant to operate for up to 12 hours with an inlet temperature up to but less than 95°F.

Date of individual notice in **Federal Register**: July 15, 1999 (64 FR 38221).

Expiration date of individual notice: August 16, 1999.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: February 26, 1999

Brief description of amendment: This amendment changes the Table Notations for Technical Specification (TS) Table 3.3–4, ''Engineered Safety Features Actuation System Instrumentation Trip Setpoints.'' Specifically, the time constants used in the lead-lag controller for Steam Line Pressure—Low (Table item 1.e) and in the rate-lag controller for Negative Steam Line Pressure Rate—High (Table item 4.e) have been revised.

Date of issuance: July 28, 1999.

Effective date: July 28, 1999.

Amendment No.: 89.

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

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14280).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 28, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application of amendments: October 2, 1998, as supplemented November 20, 1998, December 21, 1998, and May 13, 1999.

Brief description of amendments: The amendments revised the Updated Final Safety Analysis Report related to an unreviewed safety question regarding the use of a small amount of containment overpressure to ensure sufficient net positive suction head for the reactor building spray and low pressure injection pumps during the post loss of coolant accident recirculation phase.

Date of Issuance: July 19, 1999.

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

Amendment Nos.: Unit 1—305; Unit 2—305; Unit 3—305.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: June 16, 1999 (64 FR 32288).

The November 20, 1998, December 21, 1998, and May 13, 1999, letters provided clarifying information that did not change the scope of the October 2, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501

West South Broad Street, Walhalla, South Carolina.

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

Date of application for amendment: May 28, 1998.

Brief description of amendment: Changes the Crystal River Unit 3 (CR–3) licensing bases to incorporate Generic Letter 87–11, ''Relaxation in Arbitrary Intermediate Pipe Rupture Requirements,'' and NUREG/CR–2913, ''Two-Phase Jet Loads,'' as part of the licensing basis for CR–3.

Date of issuance: July 27, 1999.

Effective date: As of the date of issuance, to be incorporated into the Final Safety Analysis Report at the time of its next update.

Amendment No.: 181.

Facility Operating License No. DPR–72: Amendment approves changes to the Final Safety Analysis Report.

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38200).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of amendment request: March 1, 1999, as supplemented by letters dated March 10, 1999, June 8, 1999, and June 23, 1999.

Brief description of amendment: The amendment changes the Cooper Nuclear Station Technical Specifications to revise the calibration frequency of the reactor recirculation flow transmitters from once every 184 days to once every 18 months.

Date of issuance: July 26, 1999.

Effective date: July 26, 1999, to be implemented within 30 days.

Amendment No.: 179.

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

Date of initial notice in **Federal Register**: April 7, 1999 (64 FR 17027) The March 10, June 8, and June 23, 1999, letters provided additional clarifying information and updated TS pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305.

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

Date of amendment request: September 28, 1998, as supplemented by letter dated March 12, 1999.

Brief description of amendment: The amendment authorizes the revision to the licensing basis as described in the Updated Safety Analysis Report (USAR) to incorporate the modification for overriding the containment isolation actuation signal to the reactor coolant system letdown flow containment isolation valves.

Date of issuance: July 22, 1999.

Effective date: July 22, 1999, and shall be implemented in the next periodic update to the USAR in accordance with 10 CFR 50.71(e).

Amendment No.: 191.

Facility Operating License No. DPR–40. The amendment revised the Updated Safety Analysis Report.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64119) The March 12, 1999, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 22, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 29, 1999.

Brief description of amendment: The amendment revises Technical Specifications 2.10.2(1) and 2.10.2(3) and deletes Figure 2–11 to relocate three cycle specific parameters to the Core Operating Limits Report.

Date of issuance: July 27, 1999.

Effective date: July 27, 1999.

Amendment No.: 192.

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

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9193) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 25, 1999.

Brief description of amendments: Revise technical specifications surveillance requirement frequencies for the emergency diesel generator maintenance inspection outages, the 24-hour endurance run and the hot restart test from 18 to 24 months.

Date of issuance: July 29, 1999.

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

Amendment Nos.: 136 and 101.

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

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9196).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 2, 1999, as supplemented on April 26, 1999.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.6, ''Fuel Storage, Criticality,'' to change the maximum unirradiated fuel assembly enrichment value for new fuel storage from 4.5 to 5.0 weight percent Uranium-235 and to allow the use of equivalent criticality control to that provided by the current TS requirement of 2.35 milligrams of Boron-10 per linear inch loading in the Integral Fuel Burnable Absorber pins.

Date of issuance: July 21, 1999.

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

Amendment Nos.: 223 and 204.

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

Date of initial notice in **Federal Register**: March 10, 1999 (64 FR 11965).

The April 26, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1999.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 4th day of August 1999.

For the Nuclear Regulatory Commission.

Suzanne C. Black,

Deputy Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–20545 Filed 8–10–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Seabrook Nuclear Power Station; Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Enforcement, has issued a Director's Decision concerning a petition dated March 31, 1999, filed by Mr. David A. Lochbaum against unspecified individuals working at the Seabrook Nuclear Power Station (Seabrook Station) pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206). The petition requests that the individuals responsible for discrimination against a contract electrician at the Seabrook Nuclear Generating Station as identified in NRC Office of Investigations (OI) Report No. 1–98–005 be banned by the NRC from participation in licensed activities at and for any nuclear power plant for a period of at least five (5) years; that the individuals responsible for creating a false record to cover up the concern raised by the contract electrician as identified in the cited OI report also be banned by the NRC from participation in licensed activities at and for any nuclear power plant for a period of a least five (5) years; and that the Petitioner be permitted to attend the upcoming pre-decisional enforcement conference on this matter.

The Director, Office of Enforcement, has determined that the petition should be denied for the reasons stated in the ''Director's Decision Under 10 CFR 2.206'' (i.e., DD–99–10). While the NRC staff concluded that the foreman had engaged in wrongdoing, the Director, Office of Enforcement denied Mr. Lochbaum's request to ban the foreman from participating in licensed activities for a period of at least five years because the requested enforcement action is not appropriate based on the circumstances of the case. The Director's Decision and the Notices of Violation issued to the foreman, Williams Power Corporation, and NAESCO for the foreman's wrongdoing are available for public inspection and copying in the Commission's Public Document Room, the Gelman Building, 2120 L Street NW, Washington, DC, and on the NRC's web page at *http://www.nrc.gov/NRC/PUBLIC/2206/index.html* and *http://www.nrc.gov/OE/rpr/oehome4.htm* respectively.

A copy of the Director's Decision has been filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c). As provided therein, the Director's Decision will become the final action of the Commission twenty-five days after issuance unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland this 3rd day of August 1999.

For the Nuclear Regulatory Commission.

R. W. Borchardt,

Director, Office of Enforcement.

[FR Doc. 99–20686 Filed 8–10–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Standard Review Plan: Licensee Requests To Delay initiation of Decommissioning Activities

NRC's ''Timeliness in Decommissioning of Materials Facility'' rule (hereafter the Timeliness Rule), became effective on August 15, 1994. The Timeliness Rule established the criteria necessary to avoid future problems resulting from delayed decommissioning of contaminated inactive facilities, separate buildings, and outdoor areas.

In May 1996, the Nuclear Energy Institute (NEI) filed a petition for rulemaking to amend the Timeliness Rule to allow licensees to delay decommissioning and operate in a ''standby'' mode. NRC denied NEI's petition for rulemaking because the Timeliness Rule contains provisions which allow licensees to request delays or postponement of decommissioning, provided they can demonstrate that the delay is not detrimental to the public health and safety and is otherwise in the public interest. However, along with denying the petition, the Commission requested that NRC prepare guidance to identify the acceptance criteria necessary to demonstrate that postponement of decommissioning activities will not be detrimental to the public interest.

In response to the Commission request, NRC has developed the draft Standard Review Plan (SRP) titled, ''Licensee Requests to Delay Initiation of Decommissioning Activities.'' NRC has posted the draft SRP on the internet (www.nrc.gov/NMSS/DWM/DECOM/decomm.htm) to provide interested parties an opportunity to review and comment on NRC's acceptance criteria necessary to demonstrate that postponement of decommissioning activities will not be detrimental to the public health and safety and is otherwise in the public interest. NRC will consider all comments received in finalizing the SRP for implementation.

The draft SRP is available for inspection at the NRC's Public Document Room, 2120 L Street NW, Washington, DC 20555.

Dated at Rockville, Maryland, this 2nd day of August 1999.

For the Nuclear Regulatory Commission.

Larry W. Camper,

Chief, Decommissioning Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 99–20684 Filed 8–10–99; 8:45 am]

BILLING CODE 7590–01–M

OFFICE OF MANAGEMENT AND BUDGET

Budget Rescissions and Deferrals

TO THE CONGRESS OF THE UNITED STATES:

In accordance with the Congressional Budget and Impoundment Control Act of 1974, I herewith report one revised deferral of budget authority, now totaling $173 million.

The deferral affects programs of the Department of State.

William J. Clinton

THE WHITE HOUSE,
August 2, 1999.

Supplemental Report

Report Pursuant to Section 1013 of P.L. 93–344

This report updates Deferral No. 99–1A, which was transmitted to Congress on February 1, 1999.