would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Trojan Plant.

Agencies and Persons Consulted

In accordance with its stated policy, on July 19, 1999, the staff consulted with the Oregon State official, Mr. Adam Bless of the Oregon Office of Energy, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated August 27, 1998, 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 located at the Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, Portland, Oregon 92707.

Dated at Rockville, MD, this 16th day of August 1999.

For the Nuclear Regulatory Commission.

Louis L. Wheeler,

Acting Chief, Decommissioning Section, Project Directorate IV and Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–22030 Filed 8–24–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of August 23, 30, September 6, 13, *and* October 18, 1999.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of August 23

Tuesday, August 24

2 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)

Week of September 6—Tentative

Tuesday, September 7

9:15 a.m.—Affirmation Session (Public Meeting) (if needed)

9:20 a.m.—Briefing on PRA Implementation Plan (Public Meeting) (Contact: Tom King, 301–415–5790)

Week of September 13—Tentative

There are no meetings scheduled for the Week of September 13.

and

Week of October 18—Tentative

Thursday, October 21

9:30 a.m.—Briefing on Part 35—Rule on Medical Use of Byproduct Material (Contact: Cathy Haney, 301–415–6825) (SECY–99–201, *Draft Final Rule—10 CFR Part 35, Medical Use of Byproduct Material,* is available in the NRC Public Document Room or on NRC web site at ''www.nrc.gov/NRC/COMMISSION/SECYS/index.html''. Download the *zipped version* to obtain all attachments.)

*The schedule for Commission meetings 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 20, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–22159 Filed 8–23–99; 12:48 pm]

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 31, 1999, through August 13, 1999. The last biweekly notice was published on August 11, 1999 (64 FR 43764).

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 24, 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: August 2, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification 6.2.2.e to require either the Operations Manager or an off-shift Operations superintendent to hold a senior reactor operator (SRO) license. This revision would delete the option which allows the Manager-Operations to have at one time held a Senior Reactor Operator License for a similar unit and replaces it with the requirement for an off-shift Operations superintendent who holds an SRO license to supervise shift work and licensed activities.

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 change to Technical Specification 6.2.2.e to require the Manager-Operations or an off-shift Operations superintendent to hold an SRO license is administrative in nature and does not directly affect plant operations. The change does not physically alter the facility in any manner and, as such, does not affect the means in which any safety-related system performs its intended safety function.

Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

As stated above, the proposed change is administrative in nature. There is no physical alteration to any plant system, nor is there a change in the method in which any safety related system performs its function.

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

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

The proposed amendment does not reduce the margin of safety as defined in the Safety Analysis Report or the bases contained in the Technical Specifications. The requirement to have a licensed SRO management position responsible for plant operations is maintained within the proposed amendment. The proposed amendment is consistent with (1) 10 CFR 50.54(l), which requires individuals responsible for directing the licensed activities of licensed operators to hold an SRO license, (2) Revision 1 of NUREG–1431, ''Standard Technical Specifications Westinghouse Plants,'' and Technical Specification Traveler Form (TSTF) 65, Revision 1, and (3) the intent of ANSI/ANS–3.1, ''Standard for Selection and Training of Personnel for Nuclear Power Plants,'' (September 1979 Draft).

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 Corporate Secretary, 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: August 4, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification 6.9.1.6.2 to incorporate analytical methodology references which are used to determine core operating limits. The analytical methodologies to be referenced are documented in topical reports which have been accepted by the Nuclear Regulatory Commission for referencing in licensing applications.

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

The proposed changes incorporate additional references to methodologies used to evaluate core operating limits. These methodologies have been approved for use by the NRC. Plant structures, systems, and components will not be operated in a different manner as a result of these proposed changes and no physical modifications to equipment are involved. Adding these references to the Core Operating Limits Report section of Technical Specifications does not increase 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 changes incorporate additional references to methodologies used to evaluate core operating limits. These methodologies have been approved for use by the NRC. Plant structures, systems, and components will not be operated in a different manner as a result of these proposed changes and no physical modifications to equipment are involved. Adding these references to the Core Operating Limits Report section of Technical Specifications does not create the possibility of a new or different type of accident from any previously evaluated.

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

The proposed changes incorporate additional references to methodologies used to evaluate core operating limits. These methodologies have been approved for use by the NRC. Plant structures, systems, and components will not be operated in a different manner as a result of these proposed changes and no physical modifications to equipment are involved. Adding these references to the Core Operating Limits Report section of Technical Specifications does not involve a 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 Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Sheri R. Peterson.

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 20, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.8.A to identify the specific Containment Cooling Service Water (CCSW) equipment required to support operation of the Control Room Emergency Ventilation System (CREVS). The proposed amendment would also revise TS 3/4.5.C.2 to ensure that the suppression pool water level is adequate to prevent vortexing in the Low Pressure Coolant Injection and Core Spray pump suctions.

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

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

The proposed changes to the technical specifications provide clarity in the support system relationship and requirements for the CCSW system support of the CREVS operation. [Neither] [t]he CCSW system nor the CREVS system are assumed to be accident precursors for previously evaluated accident[s]. Therefore, the proposed changes have no effect on the probability or consequences of accidents previously evaluated.

The proposed change to the allowable suppression chamber level does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change revises a Technical Specification acceptance value to [a] more conservative value and serves to ensure operability of equipment important to safety. By ensuring equipment availability, the probability or consequences of an accident previously evaluated are not increased. In addition, the proposed changes have no impact on any initial condition assumptions for accident scenarios. Onsite or offsite dose consequences resulting from an event previously evaluated are not affected by this proposed amendment request.

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

The proposed changes do not create the possibility of a new or different kind of accident from that previously evaluated. The changes to the CCSW specifications more appropriate[ly] reflect the design requirements and clarify the support role of the CCSW system as it relates the CREVS. Neither the CCSW system nor the CREVS will be operated differently with the proposed change. Therefore new or different failure modes will not be created. Therefore, the possibility of new and different accidents has not been created with the proposed change. The proposed change to the suppression pool allowable level restores margin to the Technical Specifications and ensures equipment operability. The proposed change is conservative with respect to current requirements. The proposed amendment does not involve any plant physical changes that would create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed change to the CCSW technical specification will not result in a significant reduction in the margin of safety. The proposed change has greater consistency with the current design requirements for CSSW support of CREVS operation. Therefore, the margin of safety has been not been altered. [Therefore, the margin of safety has not been altered. SIC]

The proposed changes for suppression pool level does not involve a significant reduction in a margin of safety. In fact, the proposed changes restore margin and ensure equipment operability. Since the changes maintain the necessary level of system reliability, they do not involve a significant reduction in the margin of safety.

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

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

Date of amendment request: July 14, 1999.

Description of amendment request: The proposed amendments would allow the units to operate at an uprated power level of 3489 MWt, an increase of 5 percent rated core thermal power.

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

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

A. Evaluation of the Probability of Previously Evaluated Accidents

The proposed power uprate imposes only minor increases in plant operating conditions. No change is made to the reactor operating pressure. Operation at uprated conditions will result in moderate flow increases in those systems associated with the turbine cycle in that steam flow increases by approximately six (6)% and feed flow increases by approximately six (6)%. The increase in flow in the carbon steel piping systems was evaluated for the effect on flow induced erosion and corrosion rates and it was confirmed that power uprate has no significant effect on flow induced erosion or corrosion. The affected systems are currently monitored by the Flow Accelerated Corrosion (FAC) program that addresses erosion and corrosion concerns. Continued monitoring of the systems provides a high level of confidence in the integrity of potentially susceptible high energy piping systems.

Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, some components will be modified prior to implementation of uprated power conditions to accommodate the revised operating conditions. The review has concluded that operation at power uprate conditions will not affect the reliability of plant equipment, and that current Technical Specifications (TS) surveillance requirements ensure adequate monitoring of system operability. Systems continue to be operated in accordance with current design requirements under uprated conditions, therefore no new components or system interactions were identified that could lead to an increase in accident probability. Changes to reactor scram setpoints are such that no significant increase in scram frequency due to operation at uprated conditions will occur.

B. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences due to the Loss of Coolant Accident (LOCA) were calculated and are found to be below the applicable regulatory limits. The results are presented in Table 9–3 of Attachment E [of the July 14, 1999 submittal].

The LOCA radiological consequences have not significantly increased due to power uprate, and radiological consequences continue to meet established regulatory limits.

The radiological evaluations for other non-LOCA Design Basis Accidents (DBAs) were also performed and the dose consequences for these events did not significantly increase. These changes are outlined in Section 9.2 of Attachment E and they demonstrate that LaSalle County Station (LCS), Units 1 and 2 still meets the applicable regulatory limits.

Non-DBA Radiological Doses

All of the other radiological releases discussed in Updated Final Safety Analysis Report (UFSAR) are either unchanged because they are not power-dependent, or increase approximately in linear proportion to the amount of the uprate. The dose consequences for all of the non-LOCA radiological release accident events did not significantly increase, and are bounded by the ''LOCA Radiological Consequences'' events discussed above and were shown to meet the current dose acceptance limits. These events are discussed in Section 9.2 of Attachment E.

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

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

The configuration, operation and event response of the LCS, Units 1 and 2 systems, structures or components are [unchanged] by operation at uprated power conditions. Analysis of transient events has confirmed that the same transients remain limiting and that no transient event results in a new sequence of events that could lead to a new accident scenario.

An increase in power level will not create a new fission product release path, or result in a new fission product barrier failure mode. The current fission product barriers consisting of the reactor fuel rod cladding, the reactor coolant pressure boundary, and the containment structure remain in place. Fuel rod cladding integrity is ensured by operating within thermal, mechanical, and exposure design limits, and was confirmed for a representative core by performance of transient and accident analysis. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate the compliance with the applicable transient analysis criteria and to establish the cycle specific Minimum Critical Power Ratio (MCPR) safety limit and fuel operating limits. The integrity of the reactor coolant pressure boundary was confirmed by evaluation of the bounding overpressurization event and ensuring that the corresponding pressure remained below the American Society of Mechanical Engineers (AMSE) Boiler and Pressure Vessel (B&PV) Code, Section III, ''Rules for Construction of Nuclear Power Plant Components,'' overpressure protection requirements. Similarly, analysis of the primary containment structure has demonstrated under worst case design basis accident conditions that the containment structure remains below the containment design pressure.

The effect of operation at uprated conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at power uprated conditions does not create any new sequence of events or failure modes that lead to a new type of accident. Plant modifications required to support implementation of power uprated conditions will be made to existing systems rather than by adding new systems of a different design, which might introduce new failure modes or accident sequences.

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

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

The power uprate analysis for LCS, Units 1 and 2 assures that the power dependent safety margin will be maintained by meeting the appropriate regulatory criteria as prescribed by the applicable regulations. Similarly, factors of safety specified by application of the regulatory required design rules have been maintained, as have other acceptance criteria used to judge the acceptability of current plant operation.

No change is required in the basic duel deign to achieve the uprated power levels, or to maintain current operating and safety margins. No increase in the allowable peak bundle power is requested as a result of operation at uprated conditions. The abnormal transients have been evaluated for a representative core configuration and confirmed that operation at uprated conditions does not have an adverse effect on the operating limit MCPR. No change to the Safety Limit MCPR results, thus the margin of safety as assured by the safety limit MCPR is maintained. The fuel operating limits related to heat generation rate would still be met at uprated conditions. Cycle specific analysis will continue to be performed for each fuel reload to demonstrate the compliance with the applicable transient analysis criteria and to establish the cycle specific safety limit and fuel operating limits.

The Emergency Core Cooling System (ECCS)–LOCA performance has been evaluated at power uprated conditions using methodologies that have been approved by the NRC for 10CFR50.46, ''Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,'' analysis. The current ECCS performance requirements were used in the power uprate analysis. The ECCS–LOCA analysis was conducted at 102% of the proposed uprated thermal power in accordance with regulatory guidance. The necessary analysis for operation of General Electric (GE) fuel under uprated conditions and the determination that the peak cladding temperature (PCT) remains below the 10CFR50.46 limit of 2200°F have been performed. However, LCS Unit 2 currently contains a mixed core of GE and Siemens Power Corporation (SPC) fuel. LCS obtained [a] TS amendment that allows operation with SPC fuel, and approved the use of the SPC analytical methodology. The ECCS–LOCA analysis performed to support use of the SPC fuel was conducted at a power level that bounds 102% of the proposed uprated power level and determined that the PCT, for SPC fuel, remains below the 10CFR50.46 limit of 2200°F. The analysis for both GE and SPC fuel types demonstrate all 10CFR50.46 criteria are met. Therefore, there is no reduction in margin with respect to maintaining ECCS performance.

The margin of safety of the reactor coolant pressure boundary is maintained under power uprated conditions. The design pressure of the RPV and reactor pressure coolant pressure boundary remains at 1250 psig. The ASME B&PV Code allowable peak pressure is 1375 psig (i.e., 110% of design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a Main Steam Isolation Valve (MSIV) closure with a failure of valve position scram and this event results in a calculated peak RPV pressure of 1332 psig at the bottom of the RPV. The peak pressure remains below the 1375 psig ASME limit. Therefore, there is no decrease in margin of safety in the reactor coolant pressure boundary.

The margin of safety of the containment structure is maintained under power uprated conditions. The analyses were conducted using a newer NRC-reviewed methodology. The pre-uprated cases were run using the new methodology and the re-baselined cases were compared to the uprated cases. The short-term containment peak pressure analysis re-baseline result was 39.3 psig compared to the original analysis of 39.6 psig. At uprated conditions the peak containment drywell pressure would be 39.9 psig, and is below the design value of 45 psig. The long-term containment suppression pool temperature analysis re-baseline result was 190°F compared to the original analysis result of 200°F. At uprated conditions the analysis concluded that in the event of a LOCA, the calculated peak bulk suppression pool temperature would be 193°F. This is less than the design temperature of the suppression pool of 275°F, and the criteria used to ensure adequate Net Positive Suction Head (NPSH) to the ECCS pumps which is 212°F. Therefore, power uprate does not challenge the structural integrity of the containment structure and ECCS NPSH is assured.

Therefore, operation at power uprated conditions does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested 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: August 6, 1999

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.6.4, ''Vacuum Relief'' to remove specific operability requirements related to position indication for the suppression chamber-drywell vacuum breakers. The amendments also reformat the action statements for inoperable vacuum breakers, increase the surveillance interval for verifying that the vacuum breakers are closed, and delete the requirement to verify that the manual isolation valves are closed for an inoperable and open vacuum breaker.

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:

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

The proposed changes do not change the hardware configuration of the suppression chamber-drywell vacuum breakers, and the vacuum breakers are not considered an initiator in any accident scenario. The removal of specific indication requirements and the extension of the surveillance interval does not impact the ability of the vacuum breakers to perform their safety function. The vacuum breakers continue to meet their intended design function. The proposed changes do not impact the assumed source term for any analyzed accident. Therefore, no increases in the probability of an accident or consequences will result due to this proposed change.

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

The proposed changes do not involve any physical alterations to the suppression chamber-drywell vacuum breakers, or cause any changes in the method by which the vacuum breakers or the containment vacuum relief system performs their associated design basis functions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not impact the design function assumed for the containment vacuum relief system. The proposed changes do not require the vacuum breakers to operate in a condition not previously assumed in the facility accident analysis. The containment vacuum relief system will continue to operate and provide the protection assumed in the accident analysis. In order to limit bypass, the vacuum breakers are in a normally closed position. These vacuum breakers cannot be permanently placed in the open position. The proposed decrease in the surveillance frequency verifying the closed vacuum breakers will not increase the risk of the vacuum breakers being in the open position, since they will only open in response to a pressure differential or manual cycling. Therefore, the assurance of the operability of the containment vacuum breakers would be the same as provided under current Technical Specifications. The containment response analysis is unchanged, in that the vacuum breakers protect the containment structure, the peak containment pressure remains as calculated, and the vacuum breakers continue to maintain bypass leakage rates as assumed. Therefore this proposed change does not cause a 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.

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

Date of amendment request: July 16, 1999.

Description of amendment request: The proposed change to Technical Specification Section 3/4.7.D is to eliminate the limit for any one main steam line isolation valve (MSIV) leakage of less than or equal to 11.5 standard cubic feet per hour (scfh), and to replace that with an aggregate value of less than or equal to 46 scfh for all four MSIVs.

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 to the Technical Specifications, Appendix A, modifies the allowed leakage limit to an aggregate value with no change to the total allowed leakage rate. This change does not affect either the automatic or manual features that would close the MSIVs. There are no physical changes to the plant and plant operations remain unchanged. Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The safety function of the MSIVs is to provide a timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. The MSIVs are designed to automatically close whenever plant conditions warrant main steam line isolation. Changing the leakage limits to include an aggregate value does not affect the isolation function. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The total allowed leakage rate for all MSIVs remains unchanged at 46 scfh. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite, and, thus, the radiological analyses remain unchanged and within the guidelines of 10 CFR 100 and General Design Criteria 19. 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: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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.

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

Date of amendment request: July 27, 1999.

Description of amendment request: The proposed amendments would add a surveillance requirement to verify the Keowee out-of-tolerance logic trips and blocks closure of the appropriate overhead or underground power path breakers. This logic is being added as part of a modification to provide voltage and frequency protection for the Keowee Hydro Units to protect them from being exposed to out-of-tolerance voltage and frequency.

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

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

This change does not create any conditions or events, which lead to accidents previously, evaluated in the SAR. The Keowee Hydro units are used for mitigation of loss of power scenarios. The proposed changes do not change the current function of the Keowee Hydro Units. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated. The Keowee Hydro units and their role in the Oconee emergency power system currently meet the design/licensing basis requirements for the system. There is no adverse affect on containment integrity and no new release paths are created. The proposed changes do not cause any adverse effects to the Keowee single failure design or adversely affect the Keowee start time of 23 seconds. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The Keowee Hydro units are used for mitigation of loss of power scenarios. No accidents new or different than already evaluated in the SAR are postulated as a result of the proposed change. No setpoints for parameters, which initiate protective or mitigative action, are being changed. Therefore, this proposed amendment does not create the possibility of any new or different kind of accident.

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

The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

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

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

NRC Section Chief: Richard L. Emch, Jr.

Energy Northwest, (formerly known as the Washington Public Power Supply System), Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed amendment would change the applicability of Section 3.4.9 of the Technical Specifications (TS) from ''Mode 3 with steam drum pressure less than the RHR [residual heat removal] cut in permissive'' to ''Mode 3 with steam drum pressure less than 48 psig.'' Notes associated with TS Surveillance Requirements 3.4.9.1 and 3.5.1.2 would be changed to reflect the proposed 48 psig limit.

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

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

This change involves further restrictions on the use of RHR in the shutdown cooling mode of operation during hot shutdown conditions. Chapter 15 of the FSAR [Final Safety Analysis Report] defines the start of hot shutdown as the point when generated power is below one percent rated power. During entry into hot shutdown conditions the RHR system will be aligned in the Low Pressure Coolant Injection (LPCI) mode of operation. Thus, it will be aligned to provide water to the Reactor Pressure Vessel in the event the high pressure systems (HPCS and RCIC) are not able to perform this function. The change being proposed here has no impact on loss of coolant accidents (LOCAs) requiring mitigation using RHR aligned in the LPCI mode of operation.

During the high pressure portion of the hot shutdown condition, intersystem (LOCAs) are a concern. The purpose of the RHR SDC Isolation Reactor Pressure—High (cut-in permissive) at 135 psig is to prevent over-pressurization of portions of the RHR system. This protection is not being modified by this change. The instrumentation that provides this protection will continue to function as designed. This change only impacts the applicability of Technical Specification 3.4.9 and when RHR SDC is required to be operable.

During hot shutdown the reactor is normally cooled down through use of the main steam system and the condenser. Other means of cooling are also available using the reactor water cleanup system or a combination of emergency core cooling system (ECCS) pumps and safety relief valves (SRVs). The RHR system aligned in the SDC mode is used at the end of this cooling process to reach cold shutdown conditions of less than or equal to 200°F. The change being proposed results in the RHR SDC being manually initiated at a lower pressure and temperature. This change will have no significant impact on the capability to cool the reactor.

FSAR Chapter 15, ''Accident Analysis,'' describes two events associated with the RHR system. FSAR section 15.1.6, ''Inadvertent Residual Heat Removal Shutdown Cooling Operation,'' describes the impact of system operation during startup or cool-down when the reactor is near critical. The proposed change involves the point at which RHR is started in the SDC mode with the reactor sub-critical with control rods inserted. Therefore, there will be no change in the probability or consequences of this accident.

FSAR section 15.2.9, ''Failure of Residual Heat Removal Shutdown Cooling,'' describes the failure of the RHR system to function in SDC mode. This evaluation assumes a failure of the SDC mode of operation but does not disable the remaining modes of RHR operation. The alternate shutdown cooling paths involve the use of the SRVs [safety relief valves] to establish a cooling path to the containment suppression pool. This evaluated accident does not result in any fuel failure. The proposed change will not result in any fuel failures. The evaluated accident does result in normal coolant activity being released to the suppression pool through the safety relief valves. The proposed activity will not result in a significant change in the release of this coolant activity.

The proposed change will not cause a significant increase in the probability of a loss of SDC accident. This change proposes a delay in the use of SDC because of temperature limitations. During this time other means of decay heat removal would be used. This will result in a decrease in use of RHR in SDC mode and a decrease in the probability of failure of the system by restricting operation to be within analyzed temperature limits. The proposed change will not involve a significant increase in the consequences of the loss of shutdown cooling accident. The accident evaluated in the FSAR assumes SDC does not operate at any time and alternate means of cooling are evaluated. Section 15.2.9.6 states there is no fuel failure and release is limited to normal primary coolant activity to the suppression pool. The proposed change results in a short delay in the use of SDC because of temperature limitations. The accident described in FSAR section 15.2.9 bounds this condition and, as a result, there will be no increase in accident consequences.

With multiple means of reactor water makeup and heat removal available the restriction in the use of RHR caused by this change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will not cause any new inadvertent shutdown cooling startup, loss of water inventory or loss of cooling accidents. New or different inadvertent RHR SDC startup accidents are not possible because this change is only a further restriction on when the system is operated. The LOCA accidents during Mode 3 are bounded by the LOCAs defined for Modes 1 and 2. No new primary sytem LOCAs can be initiated because of this change. The purpose of the RHR cut-in permissive at 135 psig is to prevent overpressurization of portions of the RHR system that could cause an intersystem LOCA. This change will not result in a new or different kind of intersystem LOCA because this is only a further restriction on RHR SDC operation. The use of RHR in the SDC mode is restricted to operation at a lower pressure and temperature but other systems are available to remove the decay heat. No new or different accidents are created because of this change.

The FSAR section 15.2.9 accident, ''Failure of Resident Heat Removal Shutdown Cooling,'' is bounding for all other accidents which postulate failure of the capability to remove decay heat. No additional accidents resulting in the loss of decay heat removal capability will be caused by this change.

Therefore, the operation of WNP–2 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. The proposed change does not involve a significant reduction in a margin of safety.

The proposed amendment will increase the reliability of the RHR system when operated in shutdown cooling mode by providing assurance that the temperature limits of the piping and pipe supports will not be exceeded. The ability to protect against an intersystem LOCA is unchanged. The ability to remove decay heat from the reactor is not changed by this modification as alternate means of heat removal are available. Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a reduction in the margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

Energy Northwest (formerly known as the Washington Public Power Supply System), Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification Table 3.3.5.1–1, ''Emergency Core Cooling System (ECCS) Instrumentation Items 1.a, 2.a, 4.a and 5.a,'' to change the Reactor Vessel Water Level—Low Low Low, Level 1 allowable value from the current value of −148 inches to a new value of −142.3 inches.

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

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

This change involves the measurement of water level in the Reactor Pressure Vessel (RPV) used to initiate the ECCS. The accident evaluated for this condition is the spectrum of loss of coolant accidents (LOCA) severe enough to decrease the RPV water inventory by a significant amount.

The additional uncertainty introduced because of harsh environmental effects could not be accommodated between the existing Technical Specification allowable value and the analytical limit. This uncertainty results in a requirement that the ECCS be initiated at a slightly higher water level than previously calculated. Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will not create a new or different kind of accident since it only makes a small change in the RPV water level at which the ECCS is initiated. This change is in the conservative direction requiring a greater volume of water in the RPV to accommodate the uncertainty associated with the harsh environment of the water level sensors.

The level indicating switches are located on instrument racks in the Reactor Building. The harsh environment in this building would have no impact on the initial trip needed to initiate the ECCS on loss of RPV level since conditions in the Reactor Building would be benign at the initial stages of the accident. Only if the Level 1 trip was reset and initiated after a significant period of time would the harsh environmental conditions have an impact on the accuracy of the level indicating switches. However, increasing the water level at which the ECCS is initiated results in a more conservative value that adequately includes post-accident harsh environment uncertainties and ensures that the associated analytical limit is met.

Therefore, the operation of WNP–2 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. The proposed change does not involve a significant reduction in a margin of safety.

The proposed amendment increases the allowable value for water level in the RPV. This small increase will result in an increase in the margin of safety. A review of the plant settings for the Level 1 trip indicated that previous settings were within the new allowable value.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

Energy Northwest (formerly known as the Washington Public Power Supply System), Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement (SR) 3.5.2.2. This requirement verifies the adequacy of the water supply in the condensate storage tanks (CSTs) which support operation of the high pressure core spray (HPCS) system during Modes 4 and 5. Current Technical Specification SR 3.5.2.2 requires that CST water level be maintained above 13.25 feet in a single tank or above 7.6 feet in each tank if the suppression pool level is below its minimum level. It is proposed that the CST water level be maintained above 14.8 feet in a single tank or above 9.1 feet in each tank.

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

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

During Modes 4 and 5 HPCS may be required to provide water to the reactor vessel if the water level decreases. The revised condensate storage tank allowable levels increase the operating margins by providing an increased water inventory. The previously evaluated accident involving the loss of decay heat cooling inventory will not have an increase in probability because the inventory of water will be increased with the change being proposed.

The consequences of any accident involving the loss of decay heat cooling inventory will not change as the consequences are unaffected by the increased water inventory.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will not create a new or different kind of accident since it only increases the amount of water held in reserve to support reactor vessel inventory loss. The proposed change does not introduce any credible mechanisms for unacceptable radiation release nor does it require physical modification to the plant. The inventory of water in the CSTs will increase to support any loss of water inventory in the reactor vessel during shutdown.

The proposed change modifies the monitored values for CST level. The plant has operated well within the existing allowable values. The increased margin provided by the increased level will assure no new or different kinds of accidents result from the proposed change.

Therefore, the operation of WNP–2 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. The proposed change does not involve a significant reduction in a margin of safety.

The proposed amendment increases the allowable value for water level in the CSTs. This results in an increase in the inventory of water available for cooling and inventory control during reactor shutdown. This will result in an increase in the margin of safety.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

Energy Northwest (formerly known as the Washington Public Power Supply System), Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed amendment request would revise Technical Specification Surveillance Requirement (SR) SR 3.8.4.6 of Technical Specification 3.8.4, ''DC Sources—Operating,'' and SR 3.8.5.1 of Technical Specification 3.8.5, ''DC Sources—Shutdown.'' The proposed change to SR 3.8.4.6 would prohibit surveillance testing of Division 1, 2, and 3 125 and 250 volt DC, battery charger capacity during Modes 1, 2, and 3. However, credit could be taken for unplanned events that satisfied the surveillance requirement. The proposed change to SR 3.8.5.1 would include SR 3.8.4.6 as one of the surveillance tests that are not required to be performed.

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

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

The proposed change has no impact on previously analyzed accidents or transients, and has no effect on operation, capacity or surveillance test details of the DC system battery chargers. The change only imposes a mode restriction on performance of specified surveillance testing and allows taking credit for unplanned events that satisfy the surveillance. Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change has no effect on operation, capacity, or surveillance test details of the DC system battery chargers. The change only prohibits performing specified battery charger capacity surveillance testing from being implemented during Mode 1, 2, or 3 and allows taking credit for unplanned events that satisfy the surveillance. The proposed change to SR 3.8.4.6 of Technical Specification 3.8.4 and SR 3.8.5.1 of Technical Specification 3.8.5 are consistent with the wording previously evaluated and approved by the NRC in NUREG–1434 Rev. 1.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed change only imposes a mode restriction, prohibiting battery charger capacity surveillance testing from being performed during Modes 1, 2, and 3, allowing credit to be taken for unplanned events that satisfy the surveillance, and allowing such testing to be omitted under certain conditions during Modes 4 and 5 and during movement of irradiated fuel in secondary containment. Performance of this testing would remove a DC electrical power subsystem from service and could present a safety risk were an event to occur if the testing was performed in Modes 1, 2, and 3, or while DC service is required in other operating conditions. Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: July 20, 1998, as supplemented June 29, 1999.

Description of amendment request: The amendment would incorporate the Technical Specification changes necessary for implementation of the Boiling Water Reactor Owners' Group Reactor Stability Long-Term Solution, Enhanced Option 1–A (E1A). E1A consists of modifications to the plant operating procedures and associated plant components that provide a means for reliably detecting and avoiding reactor instabilities. By letter dated February 25, 1998, the Nuclear Regulatory Commission (NRC) staff recognized E1A as a technically acceptable implementation of a long-term stability solution satisfying the requirements of NRC IE Bulletin 88–07, Supplement 1, and Generic Letter 94–02, ''Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors.''

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

The proposed amendments allow the implementation of the Enhanced Option I–A (E1A) long term solution to the neutronic/thermal-hydraulic instability issue. Current Technical Specification (TS) restrictions on power and flow conditions, number of operating recirculation loops and operator actions implemented to reduce the probability of neutronic/thermal-hydraulic instability are eliminated and new stability requirements consistent with NEDO–32339–A, Supplement 4, Revision 1, are imposed. These requirements include restrictions on power and flow conditions and actions associated with the modified Average Power Range Monitor (APRM) flow biased scram and control rod block functions. Required actions include adherence to the boiling boundary limit stability control prior to entry and during operation in the region of the power and flow operating domain which is potentially susceptible to neutronic/thermal-hydraulic instability in the absence of the stability control. In addition, the proposed amendments require operator actions based upon control room indications generated by a new Period Based Detection System (PBDS). The PBDS is designed to provide alarm indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutronic/thermal-hydraulic instability may exist. The PBDS also provides analog indication of the highest and second highest successive period confirmation count of all of the Local Power Range Monitors (LPRMs) monitored. This provides the plant operators with continuous indication of reactor stability operating conditions.

The proposed amendments will permit operation in regions of the power and flow operating domain postulated to be susceptible to neutronic/thermal-hydraulic instability. Operation in these regions does not increase the probability of occurrence of initiators and precursors of previously analyzed accidents when neutronic/thermal-hydraulic instability is not possible. The proposed amendments permit the implementation of the features of the E1A solution which prevent neutronic/thermal-hydraulic instability including preemptive reactor scram upon entry into the regions of the power and flow operating domain most susceptible to neutronic/thermal-hydraulic instability. The E1A solution also requires implementation of stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of stability control, to neutronic/thermal-hydraulic instability. The E1A solution prevents neutronic/thermal-hydraulic instability during operation in regions of the power and flow operating domain previously excluded from operation and therefore does not significantly increase the probability of a previously analyzed accident.

Operation in the regions of the power and flow operating domain excluded by current TS 3.4.1 and Figure 3.4.1–1 can occur as a result of anticipated operational occurrences. The severity of these transients may increase in the absence of operator actions due to the potential occurrence of neutronic/thermal-hydraulic instability as a result of operation in these regions. The proposed amendments will permit the implementation of the E1A long term solution to the stability issue. Required features of the E1A solution include adherence to a boiling boundary limit stability control prior to selection by the operator of APRM flow biased scram and control rod block function ''Setup'' setpoints which allow operation in a region of the power and flow operating domain potentially susceptible, in the absence of the stability control, to neutronic/thermal-hydraulic instability. Upon entry, as a result of an anticipated operational occurrence, into the region most susceptible to neutronic/thermal-hydraulic instability, the preemptive reactor scram prevents neutronic/thermal-hydraulic instability. Therefore, the consequences of an accident do not significantly increase while operating with the stability control met.

After exiting the region requiring the stability control to be met, the setpoints can be manually reset to their normal values. Stability controls are required to be in place when setpoints are ''Setup''. As a backup E1A feature, the APRM flow biased setpoints automatically reset to their normal values above a pre-determined flow condition. This automatic reset to the more conservative setpoints ensures that the preemptive reactor scram will prevent operation as a result of an anticipated operational occurrence into the region most susceptible to neutronic/thermal-hydraulic instability should the operator not select the more conservative setpoints appropriate for operation following exit from the region requiring stability control.

Other required E1A features, including the PBDS, control rod block alarms associated with entry into the region susceptible to neutronic/thermal-hydraulic instabilities in the absence of stability controls, and required operator actions, including manual reactor scram, help ensure prevention of neutronic/thermal-hydraulic instabilities. Therefore, the proposed amendments prevent the occurrence of neutronic/thermal-hydraulic instability as a consequence of an anticipated operational occurrence and do not significantly increase the consequences of any previously analyzed accident.

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

The proposed amendments replace current restrictions on power and flow conditions with alternative restrictions which permit the implementation of the E1A long term stability solution. The current restrictions on the power and flow conditions and operating recirculation loops in the RUN mode do not automatically prevent the entry into regions of the power and flow operating domain most susceptible to neutronic/thermal-hydraulic instability and therefore the possibility of neutronic/thermal-hydraulic instability exists in the absence of operator action. The required features of the E1A solution implement a preemptive scram upon entry into the region most susceptible to neutronic/thermal-hydraulic instability, without operator action. The accessible operating domain allowed by the proposed amendments is a subset of the power and flow operating domain currently allowed. Current initiators and precursors of accidents and anticipated operational occurrences [cannot] occur with new or different initial conditions as a result of this change. Additionally, there are no new event initiators or precursors of accidents and anticipated operational occurrences created by this change. Therefore, the proposed amendments do not create the possibility of a new or different kind of accident from that previously evaluated.

Concurrent with the implementation of the proposed amendments, a modified Flow Control Trip Reference (FCTR) card, the E1A FCTR card, and a new Period Based Detection System (PBDS) will be installed as required by the E1A solution. The function of the E1A FCTR card is to aid the operator in the identification of entry into regions of the power and flow operating domain potentially susceptible to neutronic/thermal-hydraulic instability in the absence of stability controls and to initiate a preemptive scram upon entry into the regions most susceptible to neutronic/thermal-hydraulic instability. This is accomplished by altering the existing values of setpoints of the APRM flow biased scram and the control rod block functions generated by the E1A FCTR card. The E1A FCTR card design includes components which may be susceptible to electromagnetic interference or other environmental effects. The plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the environmental qualification values for the E1A FCTR cards. Therefore, the potential for spurious scrams or common mode failures induced by environmental effects (e.g., electromagnetic interference) is considered negligible. The installation of the E1A FCTR card will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated.

The function of the PBDS is to provide the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutronic/thermal-hydraulic instability may exist. This is accomplished by the installation of a new PBDS card in the Neutron Monitoring System. The PBDS card takes inputs from individual local power range monitors and provides analog indication of the highest and second highest successive period confirmation count, provides a High Decay Ratio (Hi DR) and High-High Decay Ratio (Hi-Hi DR) alarms, and INOP status indication to the operator in the control room. These displays [cannot] create the possibility of a new or different kind of accident from any accident previously evaluated. The PBDS card design includes components which may be susceptible to electromagnetic interference or other environmental effects. However, the plant specific environmental conditions (temperature, humidity, pressure, seismic, and electromagnetic compatibility) have been confirmed to be enveloped by the PBDS environmental qualification values. Therefore, the installation of the PBDS card will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. This request does not involve a significant reduction in a margin to safety.

The proposed amendments permit the implementation of the E1A long term solution to the stability issue. Under certain conditions, existing BWR [boiling water reactor] designs are susceptible to neutronic/thermal-hydraulic instability. General Design Criterion (GDC) 12 of 10 CFR 50, Appendix A, requires thermal-hydraulic instability to be prevented by design or be readily and reliably detected and suppressed. When the design of the reactor system does not prevent the occurrence of neutronic/thermal-hydraulic instability, instability is an anticipated operational occurrence. GDC 10 of 10 CFR 50, Appendix A, requires that specified acceptable fuel design limits not be exceeded during anticipated operational occurrences.

Analyses performed by the BWROG [Boiling Water Reactor Owners' Group] indicate that neutronic/thermal-hydraulic instability induced power oscillations could result in conditions exceeding the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) prior to detection and suppression by the current design of the Neutron Monitoring System and Reactor Protection System.

To ensure compliance with GDC 12 the BWROG developed Interim Corrective Actions (ICAs) to enhance the capability of the operator to readily and reliably detect and suppress neutronic/thermal-hydraulic instability. The BWROG ICAs also provided additional guidance for monitoring local power range monitors beyond the requirements of current TS 3.4.1 to ensure adequate margin to the onset of neutronic/thermal-hydraulic instability. Reliance on operator actions to comply with GDC 12 was accepted on an interim basis by the NRC pending final implementation of a long term solution to the stability issue. Neutronic/thermal-hydraulic instability is prevented by implementation of the E1A solution through the modified design of the Reactor Protection System (APRM [average power range monitor] flow biased scram) and the stability control prior to entry into a region of the power and flow operating domain which is potentially susceptible, in the absence of stability control, to neutronic/thermal-hydraulic instability. In addition, significant backup protection features, including the PBDS, control rod block alarms associated with entry into the region susceptible to neutronic/thermal-hydraulic instabilities in the absence of stability controls, and specified operator actions, including manual reactor scram, are required to be implemented. As a result, the margin to the onset of neutronic/thermal-hydraulic instability provided by the existing TS requirements and BWROG ICAs recommendations is not significantly reduced by the implementation of the E1A solution. The E1A solution assures compliance with GDC 12 by the prevention of neutronic/thermal-hydraulic instability and therefore precludes neutronic/thermal-hydraulic instability from becoming a credible consequence of an anticipated operational occurrence. The consequences of anticipated operational occurrences will not increase and the margin to the MCPR SL will not decrease upon implementation of the E1A solution. Therefore, the proposed amendments do not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 6, 1999.

Description of amendment request: The proposed amendments would change those Technical Specifications (TS) required to support Grand Gulf Nuclear Station (GGNS), Cycle 11 operation. The changes would include a change to the minimum critical power ratio safety limit (SLMCPR) that would reflect a decrease of the two recirculation loop SLMCPR limit from 1.11 to 1.09, and the single recirculation loop SLMCPR limit from 1.12 to 1.10. These values were developed with General Electric's cycle-specific SLMCPR methodology in GESTAR–II Amendment 25, which was recently approved by the Nuclear Regulatory Commission in a Safety Evaluation Report dated March 11, 1999.

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

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification 2.1.1 as that limit which ''ensures that during normal operation and during Anticipated Operational Occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling.'' The MCPR safety limit is re-evaluated for each reload and, for GGNS Cycle 11, the analyses have concluded that a two-loop MCPR safety limit of 1.09, based on the application of GE's [General Electric's] NRC-approved cycle-specific MCPR safety limit methodology demonstrates that this acceptance criterion is satisfied. For single-loop operation, a MCPR safety limit of 1.10, based on GE's [NRC-approved cycle-specific MCPR safety limit methodology, also demonstrates that this acceptance criterion is satisfied. Core MCPR operating limits are developed to support the Technical Specification 3.2 requirements and ensure these safety limits are maintained in the event of the worst-case transient. Since the MCPR safety limit will be maintained at all times, operation under the proposed changes will ensure at least 99.9% of the fuel rods in the core do not experience transition boiling. Therefore, these changes to the Minimum Critical Power Ratio (MCPR) safety limit do not affect the probability or consequences of an accident.

GE's NRC-approved GESTAR–II cycle-specific MCPR safety limit methodology has been applied and has no effect on the probability or consequences of any accidents previously evaluated. As previously licensed, one exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current licensing basis. The design of the GE11 fuel bundles is such that the bundles are not likely to be mis-oriented or mis-located and the normal administrative controls will be in effect for assuring proper orientation and location. Therefore, the probability of a fuel loading error is not increased. This analysis ensures that postulated dose releases will not exceed a small fraction (10 percent) of 10CFR100 limits. Therefore, the probability or consequences of accidents previously evaluated are unchanged.

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

The GE11 fuel to be used in Cycle 11 is of a design compatible with fuel present in the core and used in the previous cycle. Therefore, the GE11 fuel will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. The proposed revised MCPR safety limits have been shown to be acceptable for Cycle 11 operation. Compliance with the applicable criterion for incipient boiling transition continues to be ensured. The proposed MCPR safety limits do not result in the creation of any new precursors to an accident.

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

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

The MCPR safety limits have been evaluated in accordance with GE's NRC-approved cycle-specific methodology to ensure that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. One exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current licensing basis. This analysis ensures that postulated dose releases for the worst case mis-oriented and mis-located bundle will not exceed a small fraction (10 percent) of 10CFR100 limits. On this basis, the implementation of this GE methodology 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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: June 23, 1999.

Description of amendment request: The requested Technical Specification changes would revise those specifications associated with various engineered safety feature systems, which need no longer be credited following a design-basis fuel handling accident. The proposed changes affect conditions where irradiated fuel is handled in the primary or secondary containment, and also affect certain specifications related to performing core alterations. These changes are based on the revised analysis of the design-basis fuel handling accident for the Grand Gulf Nuclear Station. This requested change is consistent with the changes approved for the Perry Nuclear Power Plant Operating License (Amendment 102), and the industry-proposed change to the Technical Specification NUREGs, TSTF–51.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

A new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The proposed requirements bound the conditions of the current design basis fuel handling accident analysis which concludes that the radiological consequences are within the acceptance criteria of NUREG 0800, Section 15.7.4 and General Design Criteria 19. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

Removing a one time only allowance granted by Amendment 129 to the Operating License that is no longer in affect is an administrative change. Therefore, the proposed change does not significantly increase the probability or consequences of any previously evaluated accident.

Based on the above, neither the proposed changes to the Technical Specifications nor that to the Operating License significantly increase the probability or consequences of any accident previously evaluated.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous analyzed.

The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous analyzed.

Removing a one time only allowance granted by Amendment 129 to the Operating License that is no longer in affect is an administrative change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previous analyzed.

Based on the above, neither the proposed changes to the Technical Specifications nor that to the Operating License create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the current GGNS [Grand Gulf Nuclear Station] licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Substantial margins are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed change only eliminates the unnecessary margin from the analysis. The current margin of safety is retained.

Specifically, the margin of safety for the fuel handling accident is the difference between the 10CFR100 limits and the licensing limit defined by NUREG 0800, Section 15.7.4. With respect to the control room personnel doses, the margin of safety is the difference between the 10CFR100 limits and the licensing limit defined by 10CFR50, Appendix A, Criterion 19 (GDC 19). The additional margin between the calculated doses for the postulated events and the corresponding licensing limit provides no useful purpose.

The proposed applicability continues to ensure that the whole-body and thyroid doses at both the control room and the exclusion area and low population zone boundaries are at or below the corresponding licensing limit. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Removing a one time only allowance granted by Amendment 129 to the Operating License that is no longer in affect is an administrative change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, neither the proposed changes to the Technical Specifications nor that to the Operating License result in a significant reduction in a margin of safety.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.

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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 6, 1998.

Description of amendment request: The proposed change modifies the requirement to perform a Moderator Temperature Coefficient (MTC) test near the end of each cycle. This request constitutes a lead-plant submittal, submitted by Waterford 3 on behalf of the Combustion Engineering Owners Group (CEOG). CE NPSD–911, Amendment 1, ''Analysis of Moderator Temperature Coefficients in Support of a Change in the Technical Specifications End of Cycle Negative MTC Limit'' dated January, 1998 is provided as an Attachment to the application. Specifically, the proposed change modifies Technical Specification (TS) 4.1.1.3.2c by adding a provision that eliminates the need to determine the MTC upon reaching two-thirds of core burnup if the results of the MTC tests required in TS 4.1.1.3.2a and 4.1.1.3.2b are within a specified tolerance. In addition, some editorial changes are proposed and the Bases change is included to support the changes in the TS.

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

1. 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.

Under the proposed change, compliance with the TS Limiting Condition for Operation is achieved through a surveillance program consisting of beginning-of-cycle (BOC) measurements, plant parameter monitoring, and end-of-cycle (EOC) MTC predictions. This change eliminates the performance of the 2/3 Cycle MTC Surveillance when the BOC MTC Surveillances are within a required tolerance of the design value.

The probability and consequences of an accident previously evaluated will not be increased because this change does not modify any assumptions used in the input to the safety analyses. The current safety calculations will remain valid because the allowed range of MTC values will not change.

The Combustion Engineering analysis CE NPSD–911 and CE NPSD–911 Amendment 1, demonstrate that if the startup test program has established that the core is operating as intended, and if the isothermal temperature coefficients measured at zero power during the cycle startup program, and at power prior to 40 EFPD [Effective Full Power Days], fall within the design value of plus or minus 0.16×10^{-4} delta k/k/°F, then the end-of-cycle best estimate prediction will also be within plus or minus 0.16×10^{-4} delta k/k/°F of true MTC.

Removing the footnote that was applicable during Cycle 7 and providing a plus/minus for SR 4.1.1.3.2c is purely an administrative change.

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

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

Response: No.

Plant operation and plant parameter TS limits will remain unchanged. There are no new changes in plant design nor are any new failure modes introduced. CE NPSD–911 analysis determined that if the MTC at the beginning-of-cycle is within plus or minus 0.16×10^{-4} delta k/k/°F of the design value then the MTC at the end-of-cycle will also be within plus or minus 0.16×10^{-4} delta k/k/°F of the design value.

Removing the footnote that was applicable during Cycle 7 and providing a plus/minus for SR 4.1.1.3.2c is purely an administrative change.

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

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

Response: No.

The margin of safety will not be reduced because the range of allowed temperature coefficients will not be changed. The surveillance program consisting of beginning-of-cycle measurements, plant parameter monitoring, and end-of-cycle MTC predictions will ensure that the MTC remains within the range of acceptable values.

Removing the footnote that was applicable during Cycle 7 and providing a plus/minus for SR 4.1.1.3.2c is purely an administrative change.

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

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: July 26, 1999.

Description of amendment request: The proposed amendment would make the following line-item Technical Specification (TS) improvements:

(1) Relocate TS Section 3/4.3.3.2, Instrumentation—Incore Detectors; TS 3/4.3.3.9, Instrumentation—Waste Gas System Oxygen Monitor; and TS 3/4.4.7, Reactor Coolant System '' Chemistry, to the Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM);

(2) Change to TS 3/4.11.2, Radioactive Effluents—Explosive Gas Mixture, and TS Bases 3/4.11.2, Explosive Gas Mixture, to reflect the above proposed relocation of TS 3/4.3.3.9;

(3) Revise the requirements of TS 3/4.4.6.1, Reactor Coolant System Leakage—Leakage Detection Systems, to require one monitor (gaseous or particulate) of the containment atmosphere radioactivity monitoring systems to be operable, rather than requiring both systems to be operable simultaneously; and

(4) Revise the requirements of TS 3/4.3.3.1, Radiation Monitoring Instrumentation, to be consistent with the above proposed revision to TS 3/4.4.6.1.

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

The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiator, conditions or assumptions are affected by the proposed revisions to Technical Specification (TS) 3/4.3.3.1, Radiation Monitoring Instrumentation, TS 3/4.3.3.2, Instrumentation—Incore Detectors; TS 3/4.3.3.9, Instrumentation—Waste Gas System Oxygen Monitor; TS 3/4.4.7, Reactor Coolant System—Chemistry; TS 3/4.11.2, Radioactive Effluents—Explosive Gas Mixture; and TS 3/4.4.6.1, Reactor Coolant System Leakage—Leakage Detection Systems, and their associated TS Bases.

The requirements of TS 3/4.3.3.2, TS 3/4.3.3.9, and TS 3/4.4.7 are proposed to be relocated from the TS to the DBNPS Updated Safety Analysis Report (USAR) Technical

Requirements Manual (TRM). These requirements would be relocated generally intact to the TRM whereby future changes would be subject to the regulatory controls of 10 CFR 50.59. These relocations are consistent with the NRC guidance provided in Generic Letter (GL) 95–10, ''Relocation of Selected Technical Specifications Requirements Related to Instrumentation,'' or NUREG–1430, Revision 1, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' dated April 1995.

The proposed revision to TS 3/4.11.2, Radioactive Effluents—Explosive Gas Mixture, and its Bases is an administration change to a reference necessitated by the proposed relocation of TS 3/4.3.3.9 to the USAR TRM.

The proposed revision to TS 3/4.3.3.1 and TS 3/4.4.6.1 regarding the number of Reactor Coolant System (RCS) leakage detection monitors required and their allowed outage times is based upon the NRC's guidance of NUREG–1430, Revision 1. This proposed revision affects the TS only and does not reduce the number, diversity, or sensitivity of Reactor Coolant System leakage detection systems inside the containment building or as committed to in the DBNPS USAR.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident condition or assumption is affected by the proposed revisions. As described above, the revisions are consistent with the guidance of NRC GL 95–10 or NUREG–1430, Revision 1. The proposed revisions, as described above, do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed TS revisions. No new accident scenarios, transient precursors, failure mechanisms, or limiting failures are introduced as a result of the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed revisions do not reduce or adversely affect the capabilities of any plant structures, systems or components. The proposed relocation of TS 3/4.3.3.2, TS 3/4.3.3.9, and TS 3/4.4.7 to the USAR TRM is essentially an administrative change to the location and process by which these requirements are controlled and revised. Future revisions to these requirements relocated to the USAR TRM will be subject to the regulatory controls of 10 CFR 50.59. Therefore, these revisions will not result in a significant reduction in a margin of safety.

The proposed revision to TS 3/4.11.2 and its Bases is administrative and reflects the relocation of TS 3/4.3.3.9 to the USAR TRM. Therefore, this revision will not result in a significant reduction in a margin of safety.

The proposed revisions to TS 3/4.3.3.1 and TS 3/4.4.6.1 affect the number of containment atmosphere radioactivity monitors required by TS to be operable simultaneously. However, redundancy and diversity requirements are maintained in the TS for detecting Reactor Coolant System leakage. Although TS-allowed outage times are proposed to be increased consistent with NUREG–1430, Revision 1 guidance, related compensatory action requirements are also being increased. Furthermore, the DBNPS commitments made for complying with Regulatory Guide 1.45, May, 1973, ''Reactor Coolant Pressure Boundary Leakage Detection Systems,'' are not changed by the proposed revisions. Along with the applicable revised TS requirements, 10 CFR 50, Appendix B, Criterion XVI will require prompt corrective action for inoperable leakage detection systems. Accordingly, these proposed revisions will not result in a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: July 26, 1999.

Description of amendment request: The proposed amendment would change the Technical Specifications to adopt the performance-based 10 CFR Part 50, Appendix J, Option B approach for Type B and C containment leakage rate testing, and to relocate certain details of the tests into a Containment Leakage Testing Program.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because accident initiators, conditions, or assumptions are not affected by the proposed changes.

The proposed changes to the Technical Specifications and Bases implement 10 CFR [Part] 50 Appendix J Option B for Type B and C Local Leak Rate Testing, based on the guidance of Regulatory Guide 1.163,

''Performance-Based Containment Leak-Test Program.'' Provided that components have performed satisfactorily on a historical basis, this guidance permits the use of extended testing frequencies. These proposed changes do not affect accident initiators, conditions, or assumptions.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term or total allowable releases. With the exception of the proposed increase in the containment air lock leakage limits, the proposed changes do not affect the total allowable containment leakage rates presently specified in the Technical Specifications. Although the air lock leakage limits are proposed to be increased, the accident analyses are based on the current TS allowable maximum bypass leakage, which is not proposed to be changed. Therefore, increases in leakage limits for individual components, such as the air locks and their door seals, which are constituents of bypass leakage, will have no effect on the radiological consequences described in the accident analyses.

The proposed TS changes relating to implementation of 10 CFR [Part] 50 Appendix J Option B may result in a small, but acceptable increase in post-accident containment leakage, due to the increased probability that due to generally increased intervals between tests, an unacceptable leakage rate could go undetected for a longer length of time. NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' September, 1995, which provided the technical basis for the 10 CFR [Part] 50 Appendix J Option B rulemaking, provides a detailed evaluation of the expected leakage and its consequences and concludes that increased test frequencies are workable without significant risk impacts.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. The proposed changes do not affect the methodology used in conducting containment leak rate testing. The proposed changes do not involve a change to the plant design or operation and, therefore, will not introduce any new or different failure modes or initiators.

3. Not involve a significant reduction in a margin of safety.

The proposed changes relating to implementation of 10 CFR [Part] 50, Appendix J, Option B do not significantly affect the allowable containment leakage rates presently specified in the Technical Specifications. The Technical Specifications, under the proposed changes, will continue to ensure containment reliability by periodic testing performed in full compliance with 10 CFR [Part] 50, Appendix J.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: July 28, 1999.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.7.5.1, ''Ultimate Heat Sink,'' to allow operation on Modes 1 through 4 with an Ultimate Heat Sink water temperature of less than or equal to 90°F, instead of the current limit of less than or equal to 85°F.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are significantly affected by the proposed change. The proposed change would increase the allowable Ultimate Heat Sink (UHS) water temperature, as specified in TS LCO 3.7.5.1.b, from less than or equal to 85°F to less than or equal to 90°F. This water is used by the Service Water System to provide cooling to equipment that is used to mitigate accidents such as a Large Break Loss of Coolant Accident. This increase in Service Water temperature has been evaluated and the proposed change does not result in the operation of equipment important to safety outside their acceptable operating ranges.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not change the source term, containment isolation, or allowable releases. The proposed increase in the Service Water System temperature has been evaluated with respect to the containment and equipment used to mitigate the consequences of accidents previously evaluated. These evaluations have determined that there are no significant increases in consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed 5°F increase in UHS temperature. The proposed change does not result in installed equipment being operated outside their design operating ranges. No new or different equipment failure modes or mechanisms are introduced by the proposed change.

3. Not involve a significant reduction in a margin of safety because the proposed 5°F increase in UHS temperature does not result in significant changes to the initial conditions contributing to accident severity or consequences.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: June 17, 1999.

Description of amendment request: The proposed amendment modifies multiple surveillance requirements to support implementation of a 24-month operating cycle.

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

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

A. Frequency Extensions

The proposed Technical Specification (TS) changes involve a change in the surveillance testing intervals to facilitate a change in the Perry Nuclear Power Plant (PNPP) operating cycle from 18 months to 24 months. The proposed TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes do not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. The proposed TS changes do not impact the TS surveillance requirements themselves, or the way in which the surveillances are performed. In addition, the proposed TS changes do not introduce any accident initiators, since no accidents previously evaluated have, as their initiators, anything related to the frequency of surveillance testing. Also, evaluation of the proposed TS changes demonstrated that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident are not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, or the high reliability of the equipment. Since the impact on the systems is minimal, it is concluded that the overall impact on the plant accident analysis is negligible. Furthermore, a historical review of surveillance test results and associated maintenance records indicated that there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

B. Allowable Value Changes

The proposed changes in Allowable Values for the instrumentation include in Table 3.3.8.1–1 Items d and e of the Technical Specifications are the result of application of the Perry Instrument Setpoint Methodology (ISM) using plant specific drift values. Application of this methodology results in Allowable Values which more accurately reflect total instrumentation loop accuracy as well as that of test equipment and calculated drift between surveillances. The proposed changes will not result in any hardware changes. The instrumentation is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to these changes. The role of the instrumentation is in mitigating and thereby limiting the consequences of accidents. The Allowable Values have been developed to ensure that the design and safety analysis limits will be satisfied. The methodology used for the development of the Allowable Values ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses and that the results and radiological consequences described in the safety analyses remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

C. Frequency Reductions to Semiannual

The proposed Technical Specification (TS) changes involve a change in the surveillance testing intervals from 18 months to either 6 months or quarterly. The shorter frequencies are based on PNPP specific results of setpoint drift evaluations. The proposed more restrictive TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes do not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. The proposed TS changes do not impact the TS surveillance requirements themselves, or the way in which the surveillances are performed. In addition, the proposed TS changes do not introduce any accident initiators, since no accidents previously evaluated have, as their initiators, anything related to the frequency of surveillance testing. The proposed TS frequencies will demonstrate that the equipment and systems required to prevent or mitigate the radiological consequences of an accident are continuing to meet the assumptions of the setpoint evaluation, on a more frequent basis. Since the impact on the systems is minimal, and the assumptions of the safety analyses will be maintained, it is concluded that the overall impact on the plant accident analysis is negligible. Furthermore, a historical review of surveillance test results and associated maintenance records indicated that there was no evidence of any failures that would invalidate the proposed test frequencies. Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

A. Frequency Extensions

The proposed TS changes involve a change in the surveillance testing intervals to facilitate a change in the PNPP operating cycle length. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure modes are being introduced. In addition, the surveillance test requirements themselves, and the way surveillance tests are performed, will remain unchanged. Furthermore, a historical review of surveillance test results and associated maintenance records indicated there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

B. Allowable Value Changes

The proposed changes are the result of application of the ISM using plant specific drift values and do not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed Allowable Values does not impact safe operation of PNPP in that the safety analysis limits will be maintained. The propose Allowable Values involve no system additions or physical modifications to systems in the station. These Allowable Values were revised to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoints may be changed. Since operational methods remain unchanged and the operating parameters have been evaluated to maintain the station within existing design basis criteria, no different type of failure or accident is created.

C. Frequency Reductions to Semiannual or Quarterly

The proposed TS changes involve a change in the surveillance testing interval due to the application of the ISM and plant specific drift analysis results. Also, the quarterly tests reflect current PNPP calibration practices, since the components are normally calibrated during the Channel Functional Test. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. The proposed change does not impact core reactivity, or the manipulation of fuel bundles. As a result, no new failure modes are being introduced. In addition, the surveillance test requirements themselves, and the way surveillance tests are performed, will remain unchanged. Furthermore, a historical review of surveillance test results and associated maintenance records indicated there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

A. Frequency Extensions

Although the proposed TS changes will result in changes in the interval between surveillance tests, the impact, if any, on system availability is small, based on other, more frequent testing that is performed, or the existence of redundant systems and equipment, or overall system reliability. Evaluations have shown there is no evidence of time dependent failures that would impact the availability of the systems. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

B. Allowable Value Changes

The proposed change does not involve a reduction in a margin of safety. The proposed changes have been developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in a margin of safety.

C. Frequency Reductions to Semiannual or Quarterly

The proposed TS changes will result in a shorter interval between surveillance tests to ensure that the assumptions of the safety analysis are maintained. The impact, if any, on system availability is small, as a result of this more frequent testing that is performed. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: August 4, 1999.

Description of amendment request: The amendment would incorporate an additional option into the Required Actions for Technical Specification 3.9.1, ''Refueling Equipment Interlocks.'' The change would provide additional Required Actions when the refueling interlocks are inoperable. The alternative would permit continued refueling activities once control rod withdrawal is blocked and operators verify that all appropriate controls rods are fully inserted.

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

1. The proposed change does not involve a significant increase in the probability or

consequences of an accident previously evaluated.

The refueling interlocks are explicitly assumed in the Perry Nuclear Power Plant Updated Safety Analysis Report (USAR) analyses of the control rod removal error and fuel loading error during refueling. This analysis evaluates the probability and consequences of control rod withdrawal during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the loading of fuel, provided all required control rods are fully inserted. The refueling interlocks accomplish this by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading. When the refueling interlocks are inoperable, the current method of preventing fuel loading when a control rod is withdrawn, is to prevent fuel movement. This method is currently required by the Technical Specifications. An alternate method to ensure that fuel is not loaded into a cell with the control rod withdrawn is to prevent control rods from being withdrawn and verify that all control rods required to be inserted are fully inserted. The proposed Technical Specification Required Actions will require that a control rod block be placed in effect, thereby ensuring that control rods are not subsequently inappropriately withdrawn. Additionally, following placing the control rod withdrawal block in effect, the proposed actions will require that all required control rods be verified to be fully inserted. This verification is in addition to the requirements to periodically verify control rod position by other Technical Specification requirements. These proposed actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn, because an electrical or hydraulic block to control rod withdrawal is in place. Like the current requirements, the proposed will ensure that unacceptable operations are blocked (e.g., loading fuel into a cell with a control rod withdrawn, except when following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change). The proposed additional Required Actions provide an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn as do the current Required Action or the Surveillance Requirement. Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The change in the Technical Specification requirements does not involve a change in the plant design, or to the status of the reactor core during refueling. The proposed actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn, because an electrical or hydraulic block to control rod withdrawal is in place. Although the exact method by which the control rod withdrawal block is inserted is revised, the net effect is equivalent. The requirements will continue to ensure that fuel is not loaded into the core when a control rod is withdrawn, except when following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change. Therefore, no new failure modes are introduced, and the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As discussed in the Bases for the affected Technical Specification requirements, inadvertent criticality is prevented during the loading of fuel provided all required control rods are fully inserted during the fuel insertion. The refueling interlocks function to support the refueling procedures by preventing control rod withdrawal during fuel movement and the inadvertent loading of fuel when a control rod is withdrawn. The proposed change will allow the refueling interlocks to be inoperable and fuel movement to continue only if a control rod withdrawal block is in effect and all required control rods are verified to be fully inserted. These proposed Required Actions provide an equivalent level of protection as the refueling interlocks by preventing a configuration which could lead to an inadvertent criticality event. The refueling procedures will continue to be supported by the proposed Required Actions because control rods cannot be withdrawn and as a result fuel cannot be inadvertently loaded when a control rod is withdrawn, except when following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change. Therefore, the proposed changes do not cause 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: February 23, 1999.

Description of amendment request: The proposed license amendment would remove redundant boron concentration monitoring requirements specified for operating Modes 3 through 6 by deleting Technical Specification 3/4.1.2.9, ''Reactivity Control Systems—Boron Dilution.'' These requirements were interim measures intended to apply until a permanent boron dilution alarm system was installed and functional.

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.

The proposed amendment does not involve changes to previously evaluated accident initiators. The proposed deletion of the redundant boron concentration verification requirements do not impact the results of existing accident analyses, and will have no adverse impact on any plant system performance. TS 3/4.1.2.9 provides mode and charging pump dependent monitoring requirements for RCS boron concentration that are designed to detect an unplanned boron dilution event in MODES 3 through 6 in the absence of an automatic alarm system, and is based on the time requirements for operator action specified in Section 15.4.6 of the Standard Review Plan (SRP). This specification evolved from interim measures that were proposed by FPL until the boron dilution alarm system (BDAS) could be made completely functional following initial start-up of St. Lucie Unit 2. The BDAS is completely functional and provides redundant control room alarms to alert operators to the occurrence of an unplanned boron dilution event in Modes 3 through 6. The alarm setpoints are based on Chemical and Volume Control System (CVCS) malfunction analyses, and satisfy the same SRP acceptance criteria upon which the monitoring requirements of TS 3/4.1.2.9 were based. Therefore, operation of the facility in accordance with the proposed 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 accident previously evaluated.

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The amendment will remove requirements from the facility technical specifications that were proposed by FPL as interim measures until the boron dilution alarm system became completely functional. The amendment will not alter the design of St. Lucie plant systems described in the Updated Final Safety Analysis Report (UFSAR), and the plant configuration will continue to remain consistent with assumptions used in the existing accident analyses. Therefore, 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 accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment has been evaluated with respect to the applicable safety analyses. The BDAS provides a continuous, early warning capability to detect a boron dilution event in Modes 3, 4, 5 and 6, and satisfies the same SRP time requirements for operator action as the interim TS that is proposed for deletion. BDAS setpoints are determined and/or validated for each fuel cycle to ensure they remain consistent with the CVCS malfunction analyses of record, and changes that may become necessary are controlled pursuant to 10 CFR 50.59. The minimum required Shutdown Margin is not changed by this proposal. Therefore, operation of the facility in accordance with the proposed amendment would 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: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 27, 1999.

Description of amendment request: The proposed amendments request that Turkey Point Unit 3 Technical Specification (TS) 3/4.8.1, A.C. SOURCES,TS 3/4.4.3, PRESSURIZER, and TS 3/4.5.2, ECCS SUBSYSTEMS—T_{avg} GREATER THAN OR EQUAL TO 350°F, be revised on a one-time basis to extend the Allowed Outage Time (AOT) for an inoperable Emergency Diesel Generator (EDG) from 72 hours to 7 days. The proposed one-time AOT extension will be used to replace the Unit 3 EDG engine radiators prior to the Spring 2000 refueling outage. However, replacement of the radiator is a very labor-intensive evolution that cannot be performed within the existing 72 hour AOT. The proposed AOT extension will allow the radiator replacement activity to be completed successfully in a safe manner. The extended AOT will be applied to one EDG at a time in a sequential manner. When the radiator replacement activity is complete on one engine, it will be returned to service so that work can proceed on the redundant EDG. It should be noted that although the proposed changes apply only to Unit 3, the Unit 4 TSs are administratively affected since the TSs are combined for both units.

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

(1) 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.

The Emergency Diesel Generators (EDG) are part of the on-site electrical power distribution system. They function as a standby power source in the event that the preferred A.C. power supply, i.e., offsite power, is interrupted. While certain failures in the electrical distribution system can lead to a loss of offsite power which is a design basis event for the plant, the EDGs are not assumed to be an initiating condition of any accident evaluated in the safety analysis report. Therefore, a one-time extension in the EDG Allowed Outage Time (AOT) does not involve a significant increase in the probability of an accident previously evaluated.

The purpose of the proposed license amendment is to permit on-line replacement of the Unit 3 EDG radiators. The radiators are part of the closed-loop diesel engine cooling water system and do not interface with any system or component that contains radioactivity. The EDGs do supply A.C. power to the emergency core cooling and containment heat removal systems during accidents that involve loss of offsite power. However, no changes are predicted for the postulated post-accident releases since adequate EDG capacity will be available under the conditions of the proposed license amendment to accommodate any design basis accident condition. Accordingly, the consequences of accidents previously evaluated in the safety analysis report are not changed by an extended EDG outage.

Probabilistic Safety Assessment (PSA) techniques were used to evaluate the impact of a one-time extension of the EDG AOT from 72 hours to 7 days. The results of these analyses indicate that extending the AOT for the purpose of replacing the engine radiator cores represents an acceptably small impact on Core Damage Probability.

Based on the above, FPL concludes that the proposed amendment does not involve a significant increase in the probability or consequences of any 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.

The proposed change does not alter the design, physical configuration, or modes of operation of the plant. Plant configurations that are prohibited by Technical Specifications will not be created by the one-time EDG AOT extension. Therefore, the proposed activity does not create the possibility of 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 proposed license amendment will extend by 96 hours the requirement to shutdown the plant when a Unit 3 EDG is removed from service for maintenance. The one-time AOT extension will not alter plant equipment, setpoints, or operating practices that provide the existing margins of safety. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Sheri R. Peterson.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: August 27, 1998.

Description of amendment request: The amendment would delete the requirements for an emergency plan from the 10 CFR Part 50 license and technical specifications after the spent nuclear fuel is transferred to a Part 72 licensed independent spent fuel storage installation (ISFSI).

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

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

The proposed elimination of the emergency plan requirements from the 10 CFR 50 license is predicated on completion of transfer of the spent nuclear fuel to the proposed 10 CFR 72 ISFSI licensed area and removal of the reactor vessel and internals from the 10 CFR 50 licensed area of the site.

Removal of the potential radiological source terms for accidents previously evaluated effectively eliminates the credibility of the accidents, therefore, elimination of the emergency plan requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is deletion of emergency plan requirements and, as such, has no direct impact on plant equipment or the procedures for operating plant equipment. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Following the removal of the spent nuclear fuel and the reactor vessel and internals from the 10 CFR 50 licensed area, the remaining credible accidents are limited to decommissioning activities. The potential accidents associated with decommissioning activities are presented in the TNP [Trojan Nuclear Plant] Decommissioning Plan and have been shown to have consequences less than the EPA PAGs [Environmental Protection Agency Protective Action Guidelines]. Following the removal of the spent nuclear fuel and the reactor vessel (including the internals) from the 10 CFR 50 site, no credible accidents associated with the remaining decommissioning activities would require pre-planned emergency measures to avoid acute radiation doses. The deletion of the Trojan Nuclear Plant Permanently Defueled Emergency Plan will not result in a reduction in the margin of safety previously analyzed. Therefore, the proposed 10 CFR 50 license amendment 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: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Attorney for licensee: Leonard A. Girard, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Michael T. Masnik.

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

Date of amendment request: February 19, 1998, as supplemented July 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to revise the Radioactive Effluents Technical Specifications (RETS) in accordance with Generic Letter 89–01 (GL–89–01), to make changes to implement revised 10 CFR Part 20 requirements, and to make administrative changes under 10 CFR 50.36a.

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 license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

A. The proposed changes involve (1) combining related LCO and surveillance requirements from Sections 2.0 and 3.0, respectively, of the Indian Point 3 (IP3) RETS and relocating this text to the new Radiological Effluent Controls (REC) section of the ODCM, (2) relocating the bases contained in Section 4.0 of the RETS to the ODCM REC, (3) relocating the detailed reporting requirements contained in Section 5.0 of the RETS to the ODCM REC, and (4) updating references to 10 CFR Part 20. Additional changes include formatting both the remaining RETS and the new REG to more closely model Standard Technical Specifications (STS), revising the frequency of the Radioactive Effluent Release Report in accordance with 10 CFR 50.36a, relocating all definitions to Appendix A of the Technical Specifications and adding/deleting definitions as necessary, and adding a new Special Reports section to the ODCM. Most of the changes are (1) consistent with the guidance provided in the generic letter, NUREG–1301, or provisions of 10 CFR; or (2) editorial. Editorial changes include the relocation of text, correction of typographical and punctuation errors, renumbering, reformatting, immaterial wording revisions/deletions/clarifications which do not change intent, and updating references.

B. The proposed revisions to the liquid and gaseous release rate limits, the relocation of the old 10 CFR 20.106 requirements to the new 10 CFR 20.1302, and the revision to the TS bases for the Liquid Holdup Tank activity will involve no change in the types or amounts of effluents that will be released, nor will there be an increase in individual or cumulative occupational radiation exposures.

The changes of definitions, terminology, paragraph references, and report submittal frequency are necessary to keep IP3 TS consistent with revised federal regulations (i.e., 10 CFR 20 and 10 CFR 50.36(a)). Record retention and reporting requirements will continue to meet NRC regulations. These changes are administrative in nature and do not affect plant hardware or operation.

The changes do not impact the operation, design, configuration, or testing of plant structures, systems or components. As such, the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

A. The changes do not impact the operation, design, configuration, or testing of plant structures, systems or components. The changes do not result in a change in type or amount of radiological effluents released. As such, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

A. The changes are being made in accordance with NRC guidance and continue to assure compliance with the applicable regulatory requirements including 10 CFR 20. The changes do not result in a change in the types or amounts of effluents released. The current level of radiological effluent control will be maintained. As such, 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: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: S. Singh Bajwa.

Sacramento Municipal Utility District (the District), Docket No. 50–312, Rancho Seco Nuclear Station, Sacramento County, California

Date of amendment request: March 18, 1996 (PA–192).

Description of amendment request: The proposed amendment would update the Rancho Seco cask drop analysis and establish the cask drop event as the design-basis event for plant operation in the permanently defueled mode. The proposed amendment would also make editorial changes to the Permanently Defueled Technical Specifications and Bases by adding the word "heavy" to specification D3.3 and eliminating references to the MP–187 cask in specification D3.3 and D4.3.3.

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

The District has reviewed the proposed changes against each of the criteria in 10 CFR 50.92, and, based on the above safety analysis, concludes:

Using the Gantry Crane to handle a fully loaded transfer cask in the Fuel Storage Building will not create a significant increase in the probability or consequences of an accident previously evaluated in the SAR [safety analysis report], because the conservative dose consequence calculated for the updated, design basis cask drop event resulted in an exposure (224 mrem) that is:

1. A very small percentage ([approximately] 0.9%) of the 10 CFR 100 design basis accident dose limit of 25 rem total body;
2. A small percentage ([approximately] 3.6%) of the NUREG–0612 control of heavy loads accident dose limit of 6.25 rem total body;
3. Well within ([approximately] 4.5%) of the old EPA [Environmental Protection Agency] NUREG–0654 plume exposure Protective Action Guidelines of 500 mrem total body dose;
4. Well within the new EPA 1 to 5 rem Total Effective Dose Equivalent (TEDE) Protective Action Guidelines (PAGs) specified in document EPA–400–R–92–001, Table 2–1, May 1992;
5. Less than the maximum hypothetical Rancho Seco Independent Spent Fuel Storage Installation design basis accident (375 mrem total body dose);
6. Less than the original Rancho Seco operating design basis for the Fuel Storage Building FHA [fuel-handling accident] exposure (399 mrem);
7. Less than the original Rancho Seco operating design basis for the Reactor Building FHA exposure (477 mrem); and
8. Much less than the original Rancho Seco operating design basis Maximum Hypothetical Accident exposure (3,600 mrem).

Therefore, the conservatively calculated 224 mrem cask drop design basis accident exposure is (1) relatively small and (2) *not* considered a significant hazard.

Also, the probability of occurrence of the FHA, which is the current design basis accident, is similar to the probability of occurrence of the updated cask drop event. The FHA is assumed to occur because the fuel handling bridge is not single failure proof. Likewise for the cask drop scenario, since the Gantry Crane is not single failure proof, this Safety Analysis Report evaluates the Gantry Crane dropping a loaded spent fuel cask.

This Safety Analysis Report analyzes the dropped cask accident scenario even though the Gantry Crane and fuel handling bridge are:

1. Designed to safely handle their respective loads (i.e., a loaded transfer cask and a spent fuel assembly, respectively; and
2. In compliance with the design and administrative requirements addressed in NUREG–0612, ''Control of Heavy Loads at Nuclear Power Plants.''

A loaded cask transfer drop is a very unlikely event because of the numerous Gantry Crane safety features described in the above safety Analysis Report. These features described above include:

1. Gantry Crane Administrative Safety Features;
2. Gantry Crane Design Safety Features;
3. General Gantry Crane Control System Design Safety Features;
4. Gantry Crane Radio Control System Design Safety Features;
5. Hoist Design Safety Features; and
6. Trolly and Bridge Design Safety Features.

The updated cask drop accident scenario will not create the possibility of a new or different type of accident than previously evaluated in the SAR, because the DSAR [defueled SAR] currently evaluates a cask drop event. The cask drop scenario evaluated in the above Safety Analysis Report just updates the existing cask drop analysis. The updated cask drop analysis only:

1. Identifies the type of spent fuel cask that Rancho Seco will use;
2. Results in a change to the calculated dose consequence associated with the current, bounding, design basis accident (i.e., the FHA); and
3. Results in a change to the existing Rancho Seco cask drop analysis.

The updated, design basis, cask drop event will not involve a significant reduction in the margin of safety, because the conservatively calculated dose consequence associated with the postulated drop of a spent fuel transfer cask is:

1. Relatively small (i.e., 224 mrem) compared to the eight accident limits and previously calculated accident doses summarized above;
2. A very unlikely event;
3. Not a significant hazard; and
4. Not a public health and safety concern.

This conclusion is the same for the FHA, which is the current, bounding, Rancho Seco design basis accident.

Also, the Emergency Planning Zone remains unchanged for this updated, cask drop accident scenario. No significant changes to the Rancho Seco Emergency Plan result from this proposed change to the updated, design basis accident at Rancho Seco.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. The staff also reviewed the proposed editorial changes for no significant hazards consideration. The proposed editorial changes do not affect the design or operation of the facility and also satisfy the three standards of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the requested amendment involves no significant hazards consideration.

Local Public Document Room location: Central Library, Government Documents, 828 I Street, Sacramento, California 95814

Attorney for licensee: Dana Appling, Esq., Sacramento Municipal Utility District, P.O. Box 15830, Sacramento, California 95852–1830

NRC Section Chief: Michael T. Masnik

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

Date of amendment request: March 12, 1998, as supplemented April 24, August 20 and November 20, 1998, and February 3, 1999

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) of each unit to conform with NUREG–1431, Revision 1, ''Standard Technical Specifications—Westinghouse Plants.'' The Commission had previously issued a Notice of Consideration of Issuance of Amendments in the **Federal Register** on May 25, 1999, (64 FR 28218) covering all the proposed changes that were within the scope of NUREG–1431. The following descriptions and no significant hazard analyses cover only those items that are beyond the scope of NUREG–1431. Associated with each change are administrative/editorial changes which would make the new or revised requirements fit into the format of NUREG–1431.

1. The Standard Technical Specification (STS) terms FQW(Z) and FQC(Z) in Limiting Condition for Operation (LCO) 3.2.1 would be deleted and the terms FQ(Z), ''steady state'' limit and ''transient'' limit would be used. (Significant Hazards Evaluation A)
2. The STS wording in Required Action 3.2.4.A to ''reduce'' thermal power would be revised to ''limit'' thermal power to allow entry into the LCO applicability during startup when QPTR may be in excess of 1.02 due to transient core conditions which are usually self-correcting. (A)
3. The Applicability of LCO 3.2.4 would be revised to be consistent with the Applicability for the AFD LCO to eliminate subtle differences between the two LCO Applications which were previously the same. (M)
4. The Reactor Coolant System Loop Test specified in the TS LCO 3/4.10.4 would not be included. (L–1)
5. A new Action would be added to the Emergency Core Cooling System (ECCS)—Shutdown LCO 3.5.3. The new Action deals with the centrifugal charging subsystem. (L–2)
6. The Reactor Coolant Pump (RCP) seal injection flow requirements of 3.5.5 would be revised. The requirement to verify a single operating point would be changed to require verification of a range of values on an operating curve. (M)
7. The time allowed to reduce the power range neutron flux setpoint in 3.7.1 to within the required limit would be extended and made applicable in Mode 1 only. (L–3 and L–3a)

8. The Actions in 3.7.2 for an inoperable Main Steam Isolation Valve (MSIV) would be revised to take credit for the redundant MSIVs in each steam line. (L–4)

9. An Action would be added to the Service Water (SW) LCO 3.7.8 that accounts for the redundant automatic turbine building isolation valves in each Farley SW train. (L–5)

10. The diesel generator accelerated Test Table 3.8.1–1 would be deleted. (LA)

11. The AC Sources—Shutdown surveillance 3.8.2.1 would be revised to more clearly state the required surveillances. (L–6 and L–6a)

12. The Actions 3.8.4 and 3.8.9 for an inoperable SW intake structure Battery and Distribution System would be revised to more accurately reflect the Farley design. (L–7)

13. The STS footnote to ESFAS Table 3.3.2–1 would be revised to be consistent with the design of the Farley main steam system. (L–8)

14. A new Condition C would be added to LCO 3.3.4 to address actions associated with the source range neutron flux monitor. (M)

15. LCO 3.3.5 would be revised to accommodate the addition of a degraded grid alarm function. (M)

16. The specific title in 5.1.2 for the control room command function would be replaced with a more general description. (L–9)

17. The specific title in 5.3.1 of Health Physics Supervisor would be replaced with a more general description. (A)

18. The inspection frequency specified in 5.5.7 for the RCP flywheel would be revised to be consistent with the NRC-approved WCAP–14535A, ''Topical Report on RCP Flywheel Inspection Elimination,'' November 1996. (L–10)

19. The Health Physics Supervisor title in 5.7.1.c would be replaced with a more general description. (L–11)

20. The Emergency Diesel General (DG) Failure Report in 5.6.7 would be revised to be consistent with the latest Farley commitments for DG failure tracking and reporting. (L–12)

21. A note would be added to Surveillance Requirement (SR) 3.4.1.4 that would not require this surveillance until 7 days after reaching greater than 90% power. (M)

22. SR 3.4.5.2 would require verification that steam generator secondary side water levels are 74% (wide range). (M)

23. LCO 3.4.15 would differ from the STS in several aspects. One aspect would extend the Allowable Outage Time from 7 days to 30 days for an inoperable leakage detection system. (L–13)

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. Each proposed out-of-scope item described above is followed in parenthesis by either an A (for administrative changes), an M (for changes which would be more restrictive), an LA (for requirements that would be removed from the TS), or an L and a number (for changes that would be less restrictive). Following are the no significant hazards analyses corresponding to each of these designations.

[A—Administrative Changes]

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS. These changes involve no technical revisions to the CTS and were made to conform with the format and style of the STS. As such, these changes are administrative in nature and do not impact initiators of analyzed events or safety analyses assumptions relative to the mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. In addition, the change does not alter assumptions made in the safety analyses and licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative in nature and do not involve any technical changes. As such, these changes do not impact any safety analysis assumptions and no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

[M—More Restrictive]

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

The proposed changes provide more stringent requirements than previously existed in the CTS. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The changes are evaluated to ensure no previously analyzed accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. These changes will not alter assumptions relative to mitigation of an accident or transient event nor will they alter the operation of process variables, structures, systems, or components described in the safety analyses. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes add more restrictive requirements to the TS or make existing requirements more restrictive. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed changes do impose new or different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes add more restrictive requirements to the TS or make existing requirements more restrictive and have been evaluated to ensure consistency with the safety analysis and licensing basis. As such, these changes do not impact any safety analyses assumptions and no question of safety is involved. Therefore, these changes do not involve a reduction in a margin of safety.

[LA—Removal of Requirements]

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

The proposed changes relocate requirements from the CTS to a licensee controlled document. The document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59. Therefore, the proposed changes will only reduce the level of regulatory control on these requirements. The level of regulatory control has no impact on the probability or the consequences of an accident previously evaluated. Thus, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes relocate requirements from the CTS to a licensee controlled document. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analyses and licensing basis. Thus, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes relocate requirements from the CTS to a licensee controlled document for which future changes will be evaluated pursuant to the requirements of 10 CFR 50.59. The proposed changes do not reduce a margin of safety because they have no impact on any safety analysis assumptions. Therefore, these changes do not involve a significant reduction in a margin of safety.

[L–1—Less Restrictive]

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

The proposed change involves deleting the CTS 3/4.10.4, Reactor Coolant Loops Test Exception, requirements and does not result in any hardware changes. The proposed change deletes a test exception LCO that is no longer used or required at FNP. The natural circulation test, for which this exception is designed, was only required to be performed at FNP during the initial plant startup test program. The proposed changes do not impact the capability of the plant or any equipment to provide the required safety function as described in the FSAR. In addition, the results of the analyses described in the FSAR remain bounding. Also, the proposed changes do not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves changing the CTS requirements to delete a test exception that is no longer used and does not necessitate a physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change, which deletes CTS 3/4.10.4 does not involve a significant reduction in a margin of safety. The proposed change does not impact any safety analysis assumptions and does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, the proposed change does not impact any margin of safety.

[L–2—Less Restrictive]

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

This change does not result in any hardware changes. The ECCS components covered by this TS are not assumed to be initiators of any analyzed event. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. The change would allow the required ECCS centrifugal charging subsystem to be inoperable for up to 72 hours providing the remaining operable ECCS components can provide the flow equivalent to a single operable train which will ensure 100% of the flow assumed in the safety analyses. Since the ability of the ECCS to perform its safety function is not lost, this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will only more accurately define the minimum equipment required to be operable to perform the ECCS function while in this Condition. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change, which allows operation to continue for up to 72 hours with components inoperable in the required ECCS centrifugal charging subsystem, is acceptable based on the remaining ECCS components providing 100% of the required ECCS flow, the small probability of an event occurring in 72 hours that would require the ECCS, and the reduced potential for a unit transient resulting from the shutdown required by current TS for an inoperable required ECCS centrifugal charging subsystem. The proposed allowed outage time of 72 hours for this condition is consistent with the time currently allowed for one train of ECCS to be inoperable in Modes 1–3. The exposure of the unit to the small probability of an event requiring ECCS during this time is insignificant and offset by the benefit gained through avoiding unnecessary plant transients. Therefore, this change does not involve a significant reduction in margin of safety.

[L–3—Less Restrictive]

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

The proposed changes extend the time allowed to adjust the Power Range Neutron Flux-High trip setpoints for the case of two or more inoperable MSSVs per SG and/or positive Moderator Temperature Coefficient (MTC) and removes the requirement to adjust the Power Range Neutron Flux-High trip setpoints only one MSSV is inoperable and the MTC is zero or negative and do not result in any hardware or operating procedure changes. The affected trip setpoints, the requirement to reduce them or the time allowed to adjust them are not assumed to be an initiator of any analyzed event. In addition, the affected trip setpoints, the requirement to reduce them and the time allowed to adjust them are not a precursor to any accident analyses. Therefore, the proposed changes do not increase the probability of an accident previously evaluated. The Power Range Neutron Flux-High trip functions to mitigate the consequences of an analyzed event by shutting down the reactor. The proposed changes continue to provide assurance that the setpoints will be properly adjusted to ensure the system functions as assumed in the applicable safety analyses. Therefore, the consequences of an accident are not significantly increased.

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

The proposed changes do not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes still ensure the operability of the trip function at the correct setpoint and will facilitate the adjustment of the setpoints such that the probability of error is minimized. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The time allowed to adjust the setpoints of the affected instrumentation is not a specific assumption of any safety analysis. For the case of a single inoperable MSSV with a zero or negative MTC, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and the remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. The proposed changes still ensure the setpoints are reduced consistent with the assumptions of the safety analysis for the case of two or more inoperable MSSVs or a positive MTC. The proposed changes also reduce the potential for an inadvertent reactor trip that could result from adjusting the trip setpoints too quickly. As such, any reduction in a margin of safety will be insignificant and will likely be offset by the benefit gained from the reduced potential for an inadvertent plant trip.

[L3a—Less Restrictive]

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

The proposed change clarifies the Action requirement to reduce the power range neutron flux-high trip setpoint in Modes 2 and 3 and does not result in any hardware or operating procedure changes. The proposed change adds a note to the Action which specifies that the Action is only required in Mode 1. In Modes 2 and 3, other reactor trips (power range low and source range high) provide the required protection consistent with the acceptance criteria of the safety analysis. Therefore, the Action is not required in these Modes. The affected trip setpoints are not assumed to be an initiator of any analyzed event. In addition, the affected trip setpoints are not a precursor to any accident analyses. Therefore, the proposed change does not increase the probability of an accident previously evaluated. The affected reactor trip functions mitigate the consequences of an analyzed event by shutting down the reactor. The proposed change continues to provide assurance that the required reactor trip functions operate as assumed in the applicable safety analyses. Therefore, the consequences of an accident are not significantly increased.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change still ensures the operability of the reactor trip function at the correct setpoint for the correct Mode of operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect the ability of the MSSVs and reactor trip system to mitigate the applicable transients consistent with the assumptions of the safety analysis. The proposed change continues to ensure the acceptance criteria of the applicable safety analyses are met (primary and secondary system pressures are limited to within the required values). As such, any reduction in a margin of safety will be insignificant and will likely be offset by the benefit gained from the reduced potential for an inadvertent plant trip that could result from an error in adjusting the power range neutron flux-high trip setpoint (unnecessary in Mode 2).

[L–4—Less Restrictive]

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

The proposed change revises the Actions of the MSIV LCO in order to take credit for the redundant MSIV valves in each steam line. This change does not result in any hardware or operating procedure changes. The MSIVs are not assumed to be an initiator of any analyzed event and function to isolate the steam lines to mitigate analyzed events. As a result, the revision of this TS requirement does not affect the probability of an accident previously evaluated. The proposed change continues to provide adequate assurance that the MSIVs are either capable of performing their intended safety function or that the safety function has been performed (steam line isolated) or that power is reduced. The proposed change continues to limit plant operation when a single failure could prevent the isolation function from being accomplished. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change only affects the Actions of the MSIV LCO. The proposed change continues to ensure the MSIVs are either capable of isolating the steam lines or that the steam lines are isolated or power reduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change continues to ensure the MSIVs are either capable of isolating the steam lines or that the steam lines are isolated or power reduced. The proposed change continues to limit plant operation when a single failure could prevent the isolation function from being accomplished. Therefore, the proposed change also continues to preserve the assumptions of the applicable safety analyses. As such, the proposed change does not impact the assumptions of the applicable safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

[L–5—Less Restrictive]

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

The proposed change revises the Actions of the SWS LCO in order to take credit for the redundant automatic turbine building isolation valves in each train of SWS. This change does not result in any hardware or operating procedure changes. The turbine building isolation valves are not assumed to be an initiator of any analyzed event and function to isolate the SWS flow to non-essential components. As a result, the revision of this TS requirement does not affect the probability of an accident previously evaluated. The proposed change continues to provide adequate assurance that the turbine building isolation valves are either capable of performing their intended safety function and accommodate a single failure or that the unit is placed in a condition where the function performed by these valves is no longer required. The proposed change continues to limit plant operation when a single failure could prevent the isolation function of these valves from being accomplished. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change only affects the Actions of the SWS LCO. The proposed change continues to ensure the turbine building isolation valves are either capable of isolating the SWS system and accommodating a single failure or that the unit is placed in a condition where this isolation function is no longer required. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change continues to ensure the turbine building isolation valves are either capable of isolating the non-essential SWS loads and accommodating a single failure or that the unit is placed in a condition where the isolation function is no longer required. The proposed change continues to limit plant operation when a single failure could prevent the isolation function from being accomplished. Therefore, the proposed change also continues to preserve the assumptions of the applicable safety analyses. As such, the proposed change does not impact the assumptions of the applicable safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

[L–6—Less Restrictive]

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

The elimination of the requirement to meet surveillance tests that verify functions which are not required in the Mode of applicability of this TS will not increase the probability of any accident previously evaluated. The proposed surveillance testing continues to provide adequate assurance of the operability of the required AC Source functions and therefore, does not involve an increase in the 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?

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the operability of the required AC Source functions, continues to be determined in the same manner. Elimination of the surveillance test requirements for AC Source functions not required in these Modes does not impact the capability of the AC Sources to perform their safety function in these Modes.

[L6a—Less Restrictive]

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

The inclusion of a note consistent with the STS to provide an allowance not to perform certain surveillance tests on the AC Source required operable by the TS will not increase the probability of any accident previously evaluated. The required surveillance testing must still be performed (but not on the AC

Source while it is required operable by the TS) and will continue to provide adequate assurance of the operability of the required AC Source functions. Therefore, this change does not involve an increase in the 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?

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the operability of the required AC Source functions, continues to be determined in the same manner. The allowance not to perform certain surveillance tests on the AC Source equipment when that equipment serves to meet the TS minimum required power source ensures a stable shutdown power supply to the unit and does not impact the capability of the AC Sources to perform their safety function in these Modes.

[L–7—Less Restrictive]

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

The proposed change effectively provides a longer allowed outage time for the Service Water Intake Structure (SWIS) DC distribution and battery systems. The proposed allowed outage time is consistent with the time allowed for a Service Water train to be inoperable. The DC power sources or their associated allowed outage times are not assumed to be initiators of any analyzed event. As such, the proposed change will not increase the probability of any accident previously evaluated. The appropriate required actions consistent with that for the equipment rendered inoperable must still be performed. The proposed actions will continue to provide adequate assurance of plant safety in the same manner as if the affected equipment were inoperable for reasons other than power availability. Therefore, this change does not involve an increase in the 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?

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the inoperability of the SWIS distribution and battery systems affect only the Service Water system and the time allowed for restoration of an inoperable Service Water train remains unchanged. The allowance to declare the affected equipment inoperable and take the associated equipment TS actions continues to ensure plant safety by providing the same appropriate remedial measures for the affected equipment as would be applicable if that equipment were inoperable for reasons other than power availability. Therefore, the proposed change does not significantly impact any margin of safety.

[L–8—Less Restrictive]

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

The proposed change involves upgrading the ESFAS TS to more closely agree with the FNP design and safety analysis and does not result in any hardware changes. The proposed change revises the applicability for the initiating functions of the main steam line isolation function such that when a main steam line isolation valve is closed and the isolation function is accomplished, the automatic initiation of this function is no longer required operable. The ESFAS is not assumed to be an initiator of any analyzed event. The role of the ESFAS is in mitigating and thereby limiting the consequences of accidents. The proposed change continues to adequately ensure the operability of the ESFAS main steam line isolation function when the lines are unisolated and thereby ensures the protection provided by the function remains operable when required. Therefore, the results of the analyses described in the FSAR remain bounding. Additionally, the proposed changes do not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves upgrading the ESFAS TS to more closely agree with the FNP design and safety analysis and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety? The proposed change, which upgrades the ESFAS TS to be more consistent with the FNP design and safety analysis does not involve a significant reduction in a margin of safety. The proposed change revises the Mode of applicability for the main steam line isolation ESFAS function. The proposed change continues to adequately ensure the operability of the isolation function when it is required and thereby ensures the protection provided by the function also remains available when required. As such, the results of the analyses described in the FSAR remain bounding and this change does not have a significant impact on any design basis safety analysis.

[L–9—Less Restrictive]

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

The proposed change involves changing the CTS administrative controls requirements regarding the Shift Supervisor (SS) responsibility to more closely agree with the STS requirements and does not result in any hardware changes. The requirement to issue annual directives regarding the SS responsibilities is deleted. The title Shift Supervisor is replaced with responsible SRO. In addition, an allowance for an RO (in Modes 5 and 6) to temporarily replace the SS is added. The proposed change also eliminates the specific restriction against the STA temporarily replacing the SS. The proposed changes do not impact the capability of the plant or any equipment to provide the required safety function as described in the FSAR. In addition, the results of the analyses described in the FSAR remain bounding. Additionally, the proposed changes do not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves changing the TS administrative controls regarding the responsibilities of the SS to more closely agree with the STS requirements and eliminates the title Shift Supervisor and does not necessitate a physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes, which revise the TS administrative controls requirements for SS responsibilities to be consistent with the STS requirements and eliminate the title Shift Supervisor do not involve a significant reduction in a margin of safety. The proposed changes do not impact any safety analysis assumptions and do not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, the proposed changes do not impact any margin of safety.

[L–10—Less Restrictive]

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

The proposed change affects only the interval allowed by the TS surveillance to perform RCP flywheel inspections. The time allowed between flywheel inspections is not specifically assumed to be a precursor or initiator of any analyzed event. The studies performed to justify the proposed time interval have shown it to be adequate to detect any flaws or degradation in the RCP flywheel. As such, the proposed change does not affect the probability of any initiating events assumed in the accident analyses. The proposed change will maintain an acceptable level of safety by continuing to require RCP flywheel inspections at an interval shown to be adequate. Consequently, the proposed change will not have any affect on 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 accident previously evaluated?

The proposed change does not involve a physical alteration of the plant (no new or different types of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change only affects the interval allowed by the TS to inspect each RCP flywheel. The interval remains adequate to detect any degradation. Therefore, the possibility of a new or different kind of accident is not created by the proposed change.

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

The proposed change affects the interval allowed by the TS to inspect RCP flywheels. The proposed interval is based on the findings of WCAP–14535A and the associated NRC SER. The WCAP concludes that continued inspections of RCP flywheels are not necessary and overall plant safety could be increased by eliminating the inspections and reducing man rem dose as well as the potential for flywheel damage during disassembly and reassembly for inspection. The NRC SER requires the inspection of RCP flywheels be retained but the interval increased to once every 10 years. As such, the proposed change continues to conservatively assure the operability of the RCP flywheel while reducing man rem exposure and the potential for damage from disassembly and reassembly. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

[L–11—Less Restrictive]

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

The proposed change involves the revision of the term health physics supervisor to health physics supervision for the purpose of specifying the frequency of radiation surveillances in RWPs. The proposed change continues to provide adequate assurance that the radiation surveillances are performed within acceptable frequencies. The proposed change does not impact the capability of the plant or any equipment to provide the required safety function as described in the FSAR, or increase the potential radiation exposure of plant personnel. In addition, the results of the analyses described in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves the supervisors who specify the radiation surveillance frequencies in high radiation areas and does not necessitate a physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change, which revises the TS requirements for the personnel who specify the frequencies of radiation surveillances in high radiation areas. The proposed change allows additional supervisory personnel to specify the required frequencies. The proposed change does not impact any safety analysis assumptions and does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. In addition, the proposed change continues to ensure adequate surveillances are performed in high radiation areas. Therefore, the proposed change does not impact any margin of safety.

[L–12—Less Restrictive]

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

The proposed change involves changing the CTS administrative controls requirements regarding the Emergency Diesel Generator (EDG) failure reporting requirement and does not result in any hardware changes. The proposed change potentially reduces the number of reports received by the NRC and revises the content to include valid failures and demands. The proposed change continues to provide adequate information to assess the EDG reliability at FNP. The proposed change does not impact the capability of the plant or any equipment to provide the required safety function as described in the FSAR. In addition, the results of the analyses described in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves changing the TS administrative controls regarding the required EDG report to more closely agree with the STS requirements and does not necessitate a physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change, which revises the TS administrative controls requirement for an annual EDG report to be consistent with the STS requirement does not involve a significant reduction in a margin of safety. The proposed change does not impact any safety analysis assumptions and does not impose any new safety analyses limits or alter the plants ability to detect and mitigate events. In addition the proposed change continues to provide sufficient information to assess the reliability of the EDG at FNP. Therefore, the proposed change does not impact any margin of safety.

[L–13—Less Restrictive]

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

The proposed change extends the time allowed to restore an inoperable RCS leakage detection instrument to operable status. The CTS allow 7 days for restoration of the automatic RCS leak detection instrument and the proposed change would allow 30 days for restoration. However, adequate information continues to be furnished to the plant staff to assure that RCS leakage does not go undetected. In addition to the remaining operable automatic RCS leak detection instrument, the TS required actions provide remedial measures that ensure RCS leakage continues to be monitored by diverse means. As such, potential RCS leakage will not go undetected and operation with one required leak detection instrument inoperable continues to be limited by the TS. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any 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 introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. Therefore the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The applicable required actions and remaining operable leakage detection monitor provide adequate information to the plant staff to ensure that RCS leakage does not go undetected. In addition, operation with one required leak detection instrument inoperable continues to be limited by the TS (30 days). As such, potential RCS leakage will not go undetected and operation in the condition where a single failure could cause a loss of automatic leakage detection continues to be limited and therefore, this change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post

Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: July 29, 1999.

Description of amendment request: The proposed amendments would change the Limiting Condition for Operation 3.1.7, ''Standby Liquid Control (SLC) System.'' The proposed amendments would change ''greater than the Region B limits,'' which could be misleading, to ''within the Region B 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 proposed changes to the Unit 1 and Unit 2 Technical Specifications do not increase the probability or consequences of any previously evaluated accident or transient. These changes are administrative in nature only and are intended to revise a misleading statement in Condition A of Limiting Condition for Operation (LCO) 3.1.7, ''Standby Liquid Control (SLC) System.'' The change ensures the proper condition is entered when expected and the sodium pentaborate solution temperature, concentration, and volume limits are not exceeded without appropriate actions being taken. As currently written, Condition A of LCO 3.1.7 could be entered whenever the sodium pentaborate solution is not within Region A limits, but is greater than Region B limits as depicted in Unit 1 and Unit 2 Technical Specifications Figures 3.1.7–1 and 3.1.7–2. This is incorrect; Condition A should be entered whenever the solution is not within Region A limits, but is within Region B limits. If the solution is not within Region A limits and is greater than Region B limits, both Standby Liquid Control subsystems are inoperable and Condition C should be entered.

Technical Specifications Figure 3.1.7–1 displays the sodium pentaborate solution volume versus concentration requirements; Figure 3.1.7–2 displays the solution concentration versus temperature requirements. Each figure contains three areas: Region A, Region B, and the area not in either Region A or Region B. Region A is the permissible region of continuous operation and is represented by a four- or five-sided area. Region B is the original licensing basis region and is represented by a four-sided area. If the sodium pentaborate solution temperature, concentration, and volume combinations are within Region A, the requirements of 10 CFR 50.62, ''Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,'' are met, no condition applies, and no actions need be taken. If solution temperature, concentration, and volume combinations are not within Region A, but within Region B, then the original licensing basis is met and operation within this region is acceptable for up to 72 hours (Unit 1 FSAR, section 3.8.4, Revision 6, page 3.8–6; Unit 2 FSAR, section 4.2.3.4.3, Revision 7, page 4.2–98). If solution temperature, concentration, and volume combinations are not within either region, then the ability of the Standby Liquid Control system to shut down the reactor is not assured and only eight hours is acceptable to restore the solution to at least within Region B before the plant must be shut down.

Condition A contains misleading wording which could allow operation outside both Region A and Region B for more than eight hours. Specifically, it could be interpreted that Condition A allows the sodium pentaborate solution temperature, concentration, and volume to be greater than Region B limits for up to 72 hours. Because Region B is demarcated by a four-side area, the terms ''within Region B'' and ''greater than Region B limits'' could be interpreted to indicate different, and mutually exclusive, areas of Figures 3.1.7–1 and 3.1.7–2. Indeed, ''greater than Region B limits'' could be interpreted to refer to most or all of the area neither in Region A nor Region B. For example, 20 weight percent sodium pentaborate solution at 50°F is a point on Figure 3.1.7–2 which is ''greater than the Region B limits,'' yet it is a point at which the solution will precipitate in the storage tank rendering the system incapable of injecting the proper amount of sodium pentaborate into the reactor pressure vessel. Obviously, both Standby Liquid Control subsystems would be inoperable if the solution were at this point and Condition C should be entered to limit severely the time the unit may continue to operate with the solution in this state. However, the wording of Condition A could cause an erroneous interpretation which would inappropriately extend this time from eight to 72 hours.

The proposed changes correct the wording of Condition A to ensure this condition is not entered inappropriately and to ensure the proper condition is entered for those combinations of solution temperature, concentration, and volume not within Region A or Region B. These changes do not increase the probability of any previously evaluated accident or transient because they are administrative in nature and do not alter any plant operation or design features or requirements which could result in systems or components performing closer to their operational or design limits and thereby increasing the possibility of a failure. These changes do not increase the consequences of any previously evaluated accident or transient because they ensure the sodium pentaborate solution limits are not exceeded without appropriate actions being taken thereby ensuring the Standby Liquid Control system is capable of mitigating the consequences of an ATWS event.

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

The proposed changes to the Unit 1 and Unit 2 Technical Specifications do not create the possibility of a new or different type of accident from any previously evaluated. The changes are administrative in nature only and are intended to clarify Condition A of LCO 3.1.7. They ensure the proper condition is entered when expected and the sodium pentaborate solution temperature, concentration, and volume limits are not exceeded without appropriate actions being taken. Those limits, the conditions under which the Standby Liquid Control system is required to be operable, and the operation of the system remain unchanged and will continue to be as described, assumed, and analyzed in the Unit 1 and Unit 2 Final Safety Analysis Reports, sections 3.8 and 4.2.3.4, respectively. The only result of the proposed changes is to reduce the time limit for continued unit operation with sodium pentaborate solution temperature, concentration, or volume outside Region A and Region B from 72 hours to eight hours. Consequently, the possibility of a new or different type of accident can not be created by these changes.

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

The proposed changes to the Unit 1 and Unit 2 Technical Specifications do not involve a reduction in the margin of safety. The changes are administrative in nature only and are intended to clarify Condition A of LCO 3.1.7. They ensure the proper condition is entered when expected and the sodium pentaborate solution temperature, concentration, and volume limits are not exceeded without appropriate actions being taken. Those limits, the conditions under which the Standby Liquid Control system is required to be operable, and the operation of the system remain unchanged by the proposed changes and will continue to be as described, assumed, and analyzed in the Unit 1 and Unit 2 Final Safety Analysis Reports. Therefore, the margin of safety, that is, the ability to bring the reactor to a subcritical condition under its most reactive conditions with the Standby Liquid Control system, as embodied by the sodium pentaborate solution temperature, concentration, and volume limits and the system operability requirements will not be reduced.

In conclusion, this proposed license amendment involves no significant hazards consideration as determined by the standards set forth by the NRC in 10 CFR 50.92(c). Specifically, it has been shown in the preceding paragraphs that the proposed changes:

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

2. Do not create the possibility of a new or different type of accident from any previously evaluated, and

3. Do not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC.

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of amendment request: July 28, 1999.

Description of amendment request: The proposed amendment would add to the Technical Specifications (TS), new limiting conditions for operation and surveillance requirements for the Oscillation Power Range Monitor (OPRM) instrumentation installed in response to Generic Letter 94–02.

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:

TVA has concluded that operation of BFN [Brown Ferry Nuclear Plant] Unit 3 in accordance with the proposed change to the TS does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The proposed amendment is to enable the OPRM Upscale trip function which is contained in the previously installed PRNM [Power Range Neutron Monitoring] equipment. Enabling the OPRM hardware provides the long term stability solution required by Generic Letter 94–02. This hardware incorporates the Option III detect and suppress solution reviewed and approved by the NRC in NEDO–31960, ''BWROG Long Term Stability Solutions Licensing Methodology.'' The OPRM is designed to meet all requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic power oscillations prior to violating the fuel MCPR [minimum critical power ratio] Safety Limit. The OPRM system provides this protection in the region of the power-to-flow map where instabilities can occur, including the region where ICAs [Interim Corrective Actions] previously restricted operation because of stability concerns. Thus, the ICA restrictions on plant operations are deleted from the TS, including region avoidance and the requirement for the operator to manually scram the reactor with no recirculation loops operating. Operation at high core powers with low core flows may cause a slight, but not significant, increase in the probability that an instability can occur. This slight increase is acceptable because subsequent to the automatic detection of a design basis instability, the OPRM Upscale trip provides an automatic scram signal to the RPS which is faster protection than the operator initiated manual scram required by the current ICAs. Because of this rapid automatic action, the consequences of an instability event are not increased as a result of the installation of the OPRM system because it eliminates operator actions.

Based on the above discussion, the proposed amendment does not involve a significant increase in the probability or consequences of an 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 proposed amendment permits BFN to enable the OPRM power oscillation detect and suppress function provided in previously installed PRNM hardware, and it simultaneously deletes certain restrictions which preclude operation in regions of the power-to-flow map where oscillations potentially may occur. Enabling the OPRM Upscale trip function does not create any new system hardware interfaces nor create any new system interactions. Potential failures of the OPRM Upscale trip result either in failure to perform a mitigation action or in spurious initiation of a reactor scram. These failures would not create the possibility of a new or different kind of accident. Based on the above discussion, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The OPRM Upscale trip function implements BWROG Stability Option III, which was developed to meet the requirements of GDC 10 and GDC 12 by providing a hardware system that detects the presence of thermal-hydraulic instabilities and automatically initiates the necessary actions to suppress the oscillations prior to violating the MCPR Safety Limit. The NRC has reviewed and accepted the Option III methodology described in Licensing Topical Report NEDO–31960 and concluded this solution will provide the intended protection. Therefore, it is concluded that there will be no reduction in the margin of safety as defined in TS as a result of enabling the OPRM Upscale trip function and simultaneously removing the operating restrictions previously imposed by the ICAs.

Based on the above discussion, the proposed amendment 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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 20, 1999.

Description of amendment request: The licensee proposed the following five changes: (1) Figure 2.1–1, average power range monitor (APRM) Flow Reference Scram and APRM Rod Block Settings, the clarifying statement ''Setpoints shall be [less than or equal to] values shown on the graph'' is proposed to be added; (2) Bases Section 2.1.B, page 16, and Bases Section 3.2 APRM rod block trip discussion, page 77, the current Bases is proposed to be replaced with a more accurate discussion of the function, as identified in the Vermont Yankee Nuclear Power Station (VY) Final Safety Analysis Report (FSAR); (3) Table 3.1.1, Reactor Protection System (Scram) Instrument Requirements, APRM Upscale (Flow Bias) function, it is proposed to add ''with a maximum of 120%'' to the APRM High Flux (Flow Bias) Trip Function equation; (4) For Table 3.2.5, Control Rod-Block Instrumentation, Rod-Block Monitor (RBM) Upscale (Flow Bias) function, the caveat ''with a maximum as defined in the COLR'' [Core Operating Limits Report] is added to the Trip Setting equation; (5) For Bases page 77, it is proposed to delete the current paragraph describing the control rod-block systems and replace it with the following: ''The trip logic for the nuclear instrumentation control rod block logic is 1 out of n; i.e., any trip on one of the six APRMs, six IRMs [intermediate range monitors] or four SRMs [source range monitors] will result in a rod block. The minimum instrument channel requirements for the IRM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. The RBM is credited in the Continuous Rod Withdrawal During Power Range Operation transient for preventing excessive control rod withdrawal before the fuel cladding integrity safety limit [minimum critical power ratio] (MCPR) or the fuel rod mechanical overpower limits are exceeded. The RBM upper limit is clamped to provide protection at greater than 100% rated core flow. The clamped value is cycle specific; therefore, it is located in the Core Operating Limits Report.''

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 operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Changes 1 and 3 are administrative and have no impact on technical content; therefore, they do not increase the probability or consequences of an accident previously evaluated.

Changes 2 and 5 clarify ambiguities in the Bases. The wording is descriptive only and does not change the meaning or intent of the specification. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Change 4 adds the Rod Block Monitor Upscale (Flow Bias) maximum value limitation to the Technical Specifications. Limiting the upscale trip setting at flows in excess of 100% of rated core flow ensures the assumptions of the Continuous Rod Withdrawal During Power Range Operation Transient are met. No other accident or transient analyses are affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Change 4, limiting the maximum value for the Rod Block Monitor Upscale (Flow Bias) function, is a change to plant design, in that it clamps the upscale trip setting at flows in excess of 100% of rated core flow at the 100% core flow value. This change ensures the assumptions of the Continuous Rod Withdrawal During Power Range Operation Transient are met and has no effect on any other accident or transient analyses. Changes 1, 2, 3, and 5 do not involve a change to the plant design.

None of the proposed changes affects any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created. No safety-related equipment or safety functions, other than the Rod Block Monitor as discussed above, are altered as a result of these changes.

Based on the above VY has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Changes 1 and 3 are administrative and have no impact on technical content. Therefore, they have no effect on margin of safety.

Changes 2 and 5 clarify ambiguities in the Bases, using wording taken directly from the FSAR. The wording is descriptive only and does not change the meaning or intent of the specification. Therefore, these changes do not involve a significant reduction in a margin of safety.

Change 4 adds the Rod Block Monitor Upscale (Flow Bias) maximum value limitation to the Technical Specifications. Limiting the upscale trip setting at flows in excess of 100% of rated core flow ensures the assumptions and, therefore the margin of safety, of the Continuous Rod Withdrawal During Power Range Operation transient are met. No other accident or transient analyses are affected. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

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

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

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

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

Date of amendment request: July 30, 1999.

Brief Description of amendment: The proposed amendment would revise Required Action A.1 of Technical Specification Limiting Condition for Operation 3.7.8, ''Ultimate Heat Sink (UHS),'' to allow a Completion Time of 72 hours to restore service water temperature to less than or equal to 95°F prior to entering the required actions for plant shutdown. The amendment request was proposed as a temporary change to be in effect until September 30, 1999.

Date of publication of individual notice in the **Federal Register**: August 10, 1999 (64 FR 43406).

Expiration date of individual notice: August 24, 1999, for comments; September 8, 1999, for hearings.

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

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.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: May 23, 1997, as supplemented September 27, 1998, and May 26, 1999.

Brief description of amendments: The amendments revise the Technical Specifications to allow the installation of ABB Combustion Engineering leak tight sleeves in defective steam generator tubes as a tube repair method.

Date of issuance: August 5, 1999.

Effective date: August 5, 1999, to be implemented within 45 days.

Amendment Nos.: Unit 1—120, Unit 2—120, Unit 3—120.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 16, 1999 (64 FR 32285).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 5, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 14, 1997.

Brief description of amendments: The amendments deleted license conditions which have been satisfied, revise others to delete parts which are no longer applicable or to revise references, and make editorial changes.

Date of issuance: August 10, 1999.

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

Amendment No.: 110.

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

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19966).

The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated July 7, 1999 (64FR36722), and a Safety Evaluation dated August 10, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 30, 1999, as supplemented June 30, 1999.

Brief description of amendments: The amendments revise the Technical Specifications, Section 3/4.6.G, ''Leakage Detection Systems,'' to allow an alternate methodology for quantifying Reactor Coolant System (RCS) leakage when the normal RCS leakage detection system is inoperable.

Date of issuance: August 4, 1999.

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

Amendment Nos.: 189 & 186.

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

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24194).

The June 30, 1999, submittal provided additional clarifying information that did not change the original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 4, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 3, 1999, as supplemented May 27, and June 22, 1999.

Brief description of amendments: The amendments change the required qualifications for operations management specified in the technical specifications (TSs) for the Beaver Valley Power Station, Units 1 and 2 (BVPS–1 and BVPS–2). The requirement that the operations manager hold a Senior Reactor Operator (SRO) license at the time of appointment is changed in the TSs to require that the assistant operations managers, one for each unit, hold an SRO license on their assigned unit. The revised TSs require the operations manager to hold, or have held, an SRO license on a pressurized water reactor. Additionally, the Updated Final Safety Analysis Report (UFSAR) for each unit is changed to require the operations manager to ''hold, or have held,'' an SRO license rather than ''hold'' a license. The revised UFSARs require the same as the TSs; that the assistant operations managers hold an SRO license on the unit to which they are assigned. Finally, the amendments substitute generic personnel titles for plant-specific personnel titles in the BVPS–1 and BVPS–2 TSs. The correlation between generic titles and plant-specific titles is provided in the revised BVPS–2 UFSAR.

Date of issuance: August 10, 1999.

Effective date: Both units, as of date of issuance and shall be implemented within 60 days.

Amendment Nos.: 224 and 100.

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

Date of initial notice in **Federal Register**: April 21, 1999 (64 FR 19556).

The May 27, and June 22, 1999, letters provided additional information but did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 10, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 24, 1998, as supplemented June 23, 1999.

Brief description of amendment: The amendment approves the addition of a safety-related diesel-driven emergency feedwater pump (EFP–3) as a functional replacement for the existing motor-driven pump, addition of technical specifications and surveillances for this new pump, and deletion of cycle specific interim technical specifications which would not be required after the addition of the new pump.

Date of issuance: August 11, 1999.

Effective date: As of the date of issuance and shall be implemented prior to commencing cycle 12 operation.

Amendment No.: 182.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 13, 1999 (64 FR 2247).

The supplemental letter dated June 23, 1999, did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 11, 1999.

No significant hazards consideration comments received: No

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

No significant hazards consideration comments received: No

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

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

Date of amendment request: November 4, 1998.

Description of amendment request: To revise Technical Specifications Surveillance Requirement 4.5.2b.1 to delete the prescribed method of venting the Emergency Core Cooling System (ECCS) which would allow an alternate method to verify that the ECCS piping is full of water. In addition, the associated Bases are being revised to reflect the intent of the surveillance requirement.

Date of issuance: August 12, 1999.

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

Amendment No.: 61.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 27, 1999 (64 FR 4157)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: June 4, 1999.

Brief description of amendment: The amendment changes the Technical Specifications by extending the allowed outage time for the 32 emergency diesel generator and its fuel oil storage tank.

Date of issuance: August 9, 1999.

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

Amendment No.: 190.

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

Date of initial notice in **Federal Register:** July 6, 1999 (64 FR 36408).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 25, 1996, as supplemented April 26, 1996, September 12, 1996, March 17, 1997, September 9, 1997, December 30, 1998, and May 19, 1999.

Brief description of amendment: The amendment extends the allowed outage time for an emergency diesel generator (EDG) system from 7 to 14 days, revises requirements for EDG testing at power, and revises electrical power requirements for cold shutdown and refueling modes.

Date of issuance: July 30, 1999.

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

Amendment No.: 253.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notices in **Federal Register:** March 27, 1996 (61 FR 13532) and June 30, 1999 (64 FR 35208).

The licensee provided additional information on April 26, 1996, September 12, 1996, March 17, 1997, September 9, 1997, and December 30, 1998, that provided clarifying information within the scope of the initial **Federal Register** notice and did not change the staff's original proposed no significant hazards consideration determination. The changes proposed on May 19, 1999, were reflected in the staff's revised proposed finding of no significant hazards consideration, and encompass the additional information provided by the licensee.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 1999.

No significant hazards consideration comments received: No.

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

PP&L, Inc., Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: June 19, 1998, (Unit 1) and August 5, 1998, (Unit 2) as supplemented by letter dated November 23, 1998.

Brief description of amendments: The amendments to the Unit 1 and Unit 2 Technical Specifications (TSs) involve the addition of a new section entitled ''Oscillation Power Range Monitoring (OPRM) Instrumentation'' and revisions to Section 3.4.1 ''Recirculation Loops Operating'' to remove the specifications related to thermal power stability which are no longer required after the installation of OPRM instrumentation.

Date of issuance: July 30, 1999.

Effective date: Effective as of its date of issuance and is to be implemented within 90 days following startup from the Unit 2 ninth Refueling Inspection Outage, currently scheduled for April 16, 1999.

Amendment Nos.: 184 and 188.

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

Date of initial notice in **Federal Register:** August 12, 1998 (63 FR 43210) and August 26, 1998 (63 FR 45528).

The November 23, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 30, 1999.

No significant hazards consideration comments received: No.

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

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendments: April 23, 1999.

Brief description of amendments: The amendment changes Permanently Defueled Technical Specification D3/4.1, ''Spent Fuel Pool Level,'' to replace a specific reference to spent fuel pool (SFP) level alarm switches with a generic reference to SFP level instrumentation.

Date of issuance: August 13, 1999.

Effective date: August 13, 1999, to be implemented within 30 days.

Amendment No.: 126.

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

Date of initial notice in **Federal Register:** June 30, 1999 (64 FR 35210).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Central Library, Government Documents, 828 I Street, Sacramento, California 95814.

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

Date of amendment request: March 2, 1999, as supplemented by letter dated July 13, 1999.

Brief description of amendments: The amendments allow the use of a ''check valve with flow through the valve secured'' as an additional means to isolate an affected containment penetration (i.e., a penetration with an inoperable penetration barrier) in Technical Specification 3.6.3, Action b.

Date of issuance: August 3, 1999.

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

Amendment Nos.: Unit 1–113; Unit 2–101.

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

Date of initial notice in **Federal Register***:* April 7, 1999 (64 FR 17030).

The July 13, 1999, supplement provided additional clarifying information within the scope of the original notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 12, 1997, as supplemented by letters dated March 30, 1999, April 23, 1999, and June 18, 1999.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to extend, from 7 days to 14 days, the Allowable Outage Time applicable to an inoperable emergency diesel generator.

Date of issuance: August 2, 1999.

Effective date: August 2, 1999.

Amendment Nos.: 259 and 218.

Facility Operating License Nos. DPR–52 and DPR–68: Amendments revised the TS.

Date of initial notice in **Federal Register***:* June 30, 1999 (64 FR 35211)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1999.

No significant hazards consideration comments received: None.

Local Public Document Room location: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

Date of amendment request: May 4, 1999, as supplemented by letter dated June 4, 1999.

Brief description of amendments: The amendments correct a number of editorial errors in the Technical Specifications that occurred with the issuance of Amendment No. 64 to Facility Operating License Nos. NPF–87 and NPF–89, regarding the improved Technical Specifications conversion. In addition, Surveillance Requirement (SR) 3.8.4.7 is revised to allow the substitution of a modified performance discharge test, for a service test, for the 125 VDC batteries and SRs 3.8.1.7, 3.8.1.12, 3.8.1.15, and 3.8.1.20 are revised to separate the voltage and frequency acceptance criteria for the diesel generator start surveillances into two sets of criteria; those criteria required to be met within 10 seconds, and those criteria required to be met following achievement of steady state conditions.

Date of issuance: August 3, 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–Amendment No. 66; Unit 2–Amendment No. 66.

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

Date of initial notice in **Federal Register***:* June 2, 1999 (64 FR 29715); and June 30, 1999 (64 FR 35212).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 1, 1999, as supplemented on April 19 and April 23, 1999.

Brief description of amendment: The amendment totally replaces the current Technical Specifications Section 6.0, ''Administrative Controls.'' Administrative changes to certain other sections of the Technical Specifications were made to conform to the changes resulting from the re-write of Section 6.0.

The changes represent a comprehensive upgrade of Section 6.0 of the Vermont Yankee Technical Specifications, incorporating improvements in content and format based on industry standards. In accordance with industry practice, some Technical Specifications requirements are being relocated to the recently implemented Vermont Yankee Technical Requirements Manual, Offsite Dose Calculation Manual, or Vermont Yankee Operational Quality Assurance Manual and are being eliminated from the Technical Specification.

Date of Issuance: July 19, 1999.

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

Amendment No.: 171.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* May 19, 1999 (64 FR 27326).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 3, 1999, as supplemented by letter dated July 22, 1999.

Brief description of amendment: The amendment updates the operating license to reflect the name change of the licensee from ''Washington Public Power Supply System'' to ''Energy Northwest'' and the name change of the facility from ''WPPSS Nuclear Project No. 2'' to ''WNP–2.''

Date of issuance: August 2, 1999.

Effective date: August 2, 1999.

Amendment No.: 157.

Facility Operating License No. NPF–21: The amendment revised the operating license.

Date of initial notice in **Federal Register:** June 30, 1999. (64 FR 35214).

The July 22, 1999, supplemental letter provided additional clarifying information, did not significantly expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 29, 1999. (TSCR 211), as supplemented June 9 and July 15, 1999.

Brief description of amendments: These amendments reflect changes to Sections 15.6 and 15.7 of the Point Beach Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). The changes are considered administrative in nature and reflect personnel title changes, an increase in minimum operating crew shift staffing, relocation of the Manager's Supervisory Staff composition and functional requirements to owner-controlled documents, and revisions to the procedure review and approval process.

Date of issuance: August 11, 1999.

Effective date: August 11, 1999. The TSs shall be implemented within 90 days. Implementation also includes removal of selected requirements from TS Section 15.6, Administrative Controls, and the relocation of other requirements to licensee-controlled documents as described in the licensee's application dated January 29, 1999, as supplemented June 9 and July 15, 1999, and evaluated in the staff's safety evaluation attached to the amendments. With respect to changes to the final safety analysis report (FSAR), Wisconsin Electric Power Company shall incorporate the revisions into the next FSAR update in accordance with the schedule in 10 CFR 50.71(e).

Amendment Nos.: Unit 1–190; Unit 2–195.

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

Date of initial notice in **Federal Register**: February 24, 1999 (64 FR 9202).

The June 9 and July 15, 1999, letters provided additional clarifying information within the scope of the original **Federal Register** notice and did not affect the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 11, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

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

Date of amendment request: June 10, 1999.

Brief description of amendment: The amendment revised Technical Specification Table 3.3–4, Functional Unit 7.b., Automatic Switchover to Containment Sump (Refueling Water Storage Tank Level—Low-Low) to reflect the results of calculations that were performed for the associated instrumentation setpoints to consider the density variations due to temperature and boric acid concentrations.

Date of issuance: August 9, 1999.

Effective date: August 9, 1999, and shall be implemented within 60 days from issuance of the amendment.

Amendment No.: 126.

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

Date of initial notice in **Federal Register:** June 30, 1999 (64 FR 35215).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: June 11, 1999.

Brief description of amendment: The amendment revises TS 3.7.1.6, Steam Generator Atmospheric Relief Valves, and associated Bases to (1) require four atmospheric relief valves (ARVs) to be operable, (2) eliminate the use of ''required'' in the action statements, (3) provide action statements to address inoperability of two ARVs and three or more ARVs due to causes other than excessive leakage, and (4) limit the Limiting Condition for Operation 3.0.4 exception to when one ARV is inoperable due to causes other than excessive seat leakage.

Date of issuance: August 12, 1999.

Effective date: August 12, 1999, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 127.

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

Date of initial notice in **Federal Register:** June 30, 1999 (64 FR 35215).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1999.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 18th day of August 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–21914 Filed 8–24–99; 8:45 am]

BILLING CODE 7590–01–P