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 January 30, 1999, through February 11, 1999. The last biweekly notice was published on February 10, 1999.

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 March 26, 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, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: January 28, 1999.

Description of amendment request: The H. B. Robinson, Unit No. 2, Technical Specifications (TSs) are proposed to be changed to replace and add analytical methodologies used to determine acceptable core designs and provide inputs to methodologies that develop the core operating limits 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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes in a methodology have been previously generically reviewed and approved for use by the NRC for determining core neutronics design and gadolinimum oxide thermal conductivity. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The fuel design parameters developed in accordance with the new methodologies are bounded by the limitations in the NRC acceptance in its safety evaluations of the new methodologies. The topical reports associated with the new methodologies demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements preclude fuel rods containing gadolinium oxide from being limiting in accident and related safety analyses. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component. The proposed change will not alter the operation of any plant equipment, or otherwise increase its failure probability. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed changes to methodology continues to meet applicable design and safety analyses acceptance criteria for neutronics design analysis and gadolinimum oxide thermal conductivity. The topical reports associated with the new methodologies demonstrate that the integrity of the fuel will be maintained as is assumed or is bounded initially in accident analyses and that design requirements preclude fuel rods containing gadolinimum oxide from being limiting in accident and related safety analyses. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analyses assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

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

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

The proposed change does not involve any physical alteration of plant systems, structures, or components. The proposed changes in methodology continue to meet applicable criteria for neutronics design analysis and assure that design requirements preclude fuel rods containing gadolinimum oxide from being limiting. The proposed change does not involve a physical alteration of the plant other than allowing for fuel design in accordance with NRC approved methodologies. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change is to methodologies that continue to meet applicable criteria for neutronics design analysis and continues to assure that design requirements preclude fuel rods containing gadolinimum oxide from being limiting. The proposed change does not impact the condition or performance of structures, systems, setpoints, 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 result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Cecil B. Thomas.

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

Date of amendment request: November 25, 1998.

Description of amendment request: The proposed amendments would revise Improved Technical Specifications 3.8.4 and 3.8.9 to support on-line replacement of the Braidwood 125 Volt DC AT&T batteries with new Charter Systems Inc. batteries.

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?

During the replacement of the existing batteries, a temporary battery bank will provide the same function as the AT&T batteries being removed. Even though this temporary battery will not be seismically mounted, due to its location in the Turbine Building, it is the safety related AT&T battery which was previously qualified and used to perform this function on Unit 1.

While the temporary battery is being connected, the DC bus will be supplied by the existing crosstie with Unit 1. Similar crosstie conditions are allowed under the present Improved Technical Specifications.

The DC system is normally supplied by the AC system through the ESF [Engineered Safety Feature] battery charger. The essential function of the DC system battery is to supply control power necessary to start and load the Diesel Generators. Once the Diesel Generators are on line, the DC system will be supplied via the battery charger. However, the ESF batteries have been sized for one hour to provide additional assurance that the critical DC loads are available in the event of a loss of a battery charger.

During the 10 day Completion Time when the temporary battery and the ESF charger are supporting the bus, the ability of that DC Division to mitigate an event/accident is unchanged except for its ability to cope with a seismic event. However, the probability of a seismic event concurrent with the 10 day Completion Time is extremely small. During a seismic event, one DC division may be compromised, however, the unit has adequate DC power available in the form of the other division to mitigate all Design Basis accidents. This loss of one DC division is bounded by the loss of an entire AC division, a condition which the plant is currently evaluated to withstand.

During the 8 hour Completion Time to connect and disconnect the temporary battery, there is no adverse impact on Unit 1. The compensatory measures to manually open the crosstie will ensure the Unit 1 DC battery can supply its required loads for the entire one hour duty cycle. The Unit 2 DC bus, which is crosstied, will be de-energized in the event of a Unit 2 accident based on the compensatory measures. This action would only be required if the associated Diesel Generator were to fail to re-energize its associated charger. This condition is consistent with the other crosstie scenarios currently permitted by the Technical Specifications. Thus, the 8 hour Completion Time is consistent with the two hour Completion Time with respect to the ability to safely shutdown the Unit. Only the duration of the Completion Time is different.

Based on the above, the overall design, function, and operation of the DC system and equipment has not been significantly modified by these changes. The proposed changes do not affect any accident initiators or precursors and do not alter the design assumptions for the systems or components used to mitigate the consequences of an accident as analyzed in UFSAR Chapter 15.

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?

During the replacement of the existing batteries, a temporary battery bank will provide the same function as the batteries being removed. Even though this temporary battery is not seismically mounted, it is the safety related AT&T battery which was previously qualified and utilized to perform this function on Unit 1. Because this temporary battery is identical to the battery that is currently installed, and will be connected and used in the same way, no new electrical or functional failure modes are created.

The temporary battery will be located in the turbine building, which is non-seismic. The temporary battery will not be seismically mounted. Thus, a seismic failure of the batteries is possible. Since the temporary battery is located in the turbine building the potential for battery failure to initiate an accident is not present, and failure of the battery cannot create a different response from any previously postulated accident.

Due to the location of the main generator in relationship to the temporary batteries, a turbine blade failure would not hit the battery unless it penetrated the turbine casing and ricocheted in the direction of the battery, which is an unlikely scenario due to the orientation of the temporary battery. Likewise, an unmitigated Outside Containment Steam Line Break of either unit would be interrupted by the successful closure of all MSIVs [Main Steam Isolation Valves] thereby leaving the battery and the DC bus intact and available. Also any affects of a postulated storm on the turbine building have been previously addressed and would not change as a result of the batteries being temporary located there.

While the temporary battery is being connected, the DC bus will be supplied by the existing crosstie with Unit 1. To prevent any occurrence on Unit 2 from adversely affecting Unit 1, this crosstie will be manually disconnected based on specific criteria that may be indicative of a Unit 2 accident (specifically a Unit 2 LOOP). Once the crosstie is opened, the Unit 2 bus will be de-energized and the other Unit 2 division will be required to mitigate the accident. This loss of one DC division is bounded by the loss of one division (AC or DC), a condition which the plant is currently evaluated to withstand.

The DC system and its equipment will continue to perform the same function and be operated in the same fashion. The proposed changes do not introduce any new accident initiators or precursors, or any new design assumptions for the systems or components used to mitigate the consequences of an accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

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

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

During the replacement of the existing batteries, a temporary safety related battery bank will perform the same function as the batteries being removed. Even though this temporary battery is not seismically mounted, it is the safety related battery which was previously qualified and used to perform this function on Unit 1 and is identical to the safety related battery that is currently installed. Therefore, it has the same capacity, margin and capability to fulfill the requirements of the Unit 2 DC bus as the existing qualified battery. The proposed replacement activity will not prevent the plant from responding to either a seismic event or design basis accident. In both cases, the design mitigation capability will be maintained. Due to the limited duration of the activity and the planned contingency actions, a significant reduction in the margin of safety will not result.

While the temporary battery is being connected, the DC bus will be supplied by the existing crosstie with Unit 1. This condition is currently allowed for a limited time by the Improved Technical Specifications.

The inherent design conservatism of the DC system and its equipment has not been altered. The DC system and its equipment will continue to be operated with the same degree of conservatism. Accordingly, there is no significant reduction in the margin of safety.

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

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

NRC Project Director: Stuart A. Richards.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: December 29, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specification Tables 3.3.1–1 and 3.3.2–1, to revise twelve Reactor Trip System and Engineered Safety Feature Actuation System Allowable Values.

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

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

These changes to the twelve AVs [Allowable Values] do not involve an increase in the probability of an accident previously evaluated. The AVs provide the basis for determining instrument channel operability and do not change the system function, or channel operation or calibration. Operation within the AV ensures the instrument channel's ability to provide the required reactor trip or engineered safety feature actuation signal during plant operation. In all cases, the proposed changes only make the twelve AVs more restrictive with respect to the current AVs, and do not effect the response characteristics of the instrumentation because actual trip setpoints are unchanged. There is no change being made to the approved design, nor is there any operational change being made which would increase the probability of occurrence of an accident previously evaluated. The RTS [Reactor Trip System] and ESFAS [Engineered Safety Feature Actuation System] systems which are actuated by the corresponding instrumentation setpoints will operate in the same manner as before and within their design limits.

These changes to the twelve AVs do not involve an increase in the consequences of an accident previously evaluated. These changes have no effect on plant operation. There is no physical or operational change being made which would alter the sequence of events, plant response, or assumptions or conclusions of the affected analyses. The use of the AVs as a basis for determining instrument or channel operability does not change system operation or channel function. The proposed changes do not change the established trip setpoints for these functions. No design analyses have changed or will be affected. The twelve revised AVs are more restrictive than the current AVs and continue to ensure that the safety limits are not violated during anticipated transients, and that the consequences of design basis accidents remain acceptable. The change to the AVs does not degrade or prevent any actions from taking place in response to an accident. The use of NRC approved or endorsed methodology in developing the proposed AVs ensures that the present analytical limits for all accidents will be maintained. These proposed changes to the AVs for RTS and ESFAS instrumentation will continue to ensure that the associated RTS trip or ESFAS actuation signals will be generated when required within the bounds of the plant safety analyses. There is no change in the type or amount of any effluents released, and no change in either the onsite or offsite dose consequences as a result of this change.

Therefore, based on this evaluation, this proposed amendment 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?

These proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the twelve AVs for RTS and ESFAS instrumentation will not affect the trip setpoints at which a reactor trip or engineered safety feature actuation is initiated. The trip setpoints contained in the Technical Requirements Manual are not being changed and will continue to be maintained. The only changes being made are to the AVs used as a basis for determining instrument channel operability. Because the trip setpoints are unchanged, RTS or ESFAS setpoint actuation is not affected by the revised AVs.

An RTS trip or ESFAS actuation signal that may initiate between its trip setpoint and the associated AV is acceptable because an allowance has been made in the affected instrument uncertainty calculation to accommodate this deviation. It allows for potential drift while ensuring plant operation in a safe manner. Using this methodology provides plant operational flexibility and yet remains within the allowances accounted for in the various accident analyses. No new equipment is being installed, and no installed equipment is being operated in a new or different manner with these twelve AV changes. The revised AVs do not alter the intended design or operation of systems or instrument channels.

As no physical plant equipment changes are being made, no new equipment failure modes are being introduced as a result of these proposed changes. There is no change in plant operation that affects previously evaluated failure modes and no change in plant response to a transient condition. These changes do not represent a new failure mode over what has been previously evaluated.

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

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

There is no significant reduction in the margin of safety from these proposed changes. These proposed changes move twelve AVs closer to the trip setpoints compared to the existing AVs, which increases the margin of safety. An RTS trip or ESFAS actuation signal that may initiate between its trip setpoint and the associated AV is acceptable because an allowance has been made in the affected instrument uncertainty calculation to accommodate this deviation. The revised AVs have been calculated using NRC approved or endorsed methodology, which is consistent with existing safety analyses that define the margin of safety. Safety analyses assumptions and results are not affected.

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

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

NRC Project Director: Stuart A. Richards.

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: January 21, 1999.

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

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

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

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

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

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

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

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

The proposed amendment represents the relocation of current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed changes are administrative in nature and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Quad Cities Nuclear Power Station. Since the proposed changes are administrative in nature, and are based on NRC accepted provisions which have been adopted at other nuclear facilities, and maintain the necessary levels of system reliability, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: 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 Project Director: Stuart A. Richards.

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

Date of amendment request: January 28, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to correct Surveillance Requirement (SR) 3.7.13.4 and the associated Bases. This SR currently is incorrect and does not reflect the Fuel Handling Ventilation Exhaust System (FHVES) as designed. Specifically, the FHVES flow rate requirement has been inadvertently stated at half the design value (18,221 instead of 36,443 cfm [cubic feet per minute]). The proposed amendments would only revise the SR to the correct design value; no physical change to the FHVES design is involved.

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

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no effect on accident probabilities or consequences. The FHVES is not an accident initiating system; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the system is not being modified by this proposed amendment. The amendment merely aligns TS requirements with the existing design and function of the system. Therefore, there will be no impact on any accident consequences.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; neither does it impact any accident mitigating systems.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. The FHVES is already capable of performing as designed. No safety margins will be impacted.

Based upon the preceding analysis, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

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

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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: December 16, 1998, supplemented January 25, 1999.

Description of amendment request: The proposed amendments would completely replace the High Pressure Injection (HPI) section of the Improved Technical Specifications that were issued on December 16, 1998. The proposed changes would: (1) expand the applicability for the requirements regarding the third HPI pump, discharge crossover valves, and the HPI suction headers; (2) specify the HPI conditions and allowed times that require the discharge headers be cross-connected or separated; (3) incorporate limiting conditions for operation when specified equipment was inoperable during specified plant conditions; (4) specify changes in HPI system discharge path valve lineup when certain equipment is inoperable; (5) change the requirement to reduce reactor power when an HPI system is inoperable from 60 percent power to 75 percent power and specify the length of time operation may continue at this power level; (6) address the failure to cross-connect the HPI discharge headers as an independent condition; (7) add a requirement to verify by administrative means that the Atmospheric Dump Valve flow path for each steam generator is operable every 12 hours under certain conditions; (8) add a requirement that the HPI pump and crossover valves be restored to operable status within 30 days; (9) delete the requirement to restore the capability to automatically actuate the HPI within 24 hours; (10) add a Required Action to reduce reactor power to less than or equal to 75 percent power within 3 hours in the event an HPI train cannot be actuated by automatic or manual means; (11) expand the Completion Time for restoring an inoperable HPI train to 72 hours; (12) require that Limiting Condition for Operation 3.0.3 be entered immediately if two HPI trains or two HPI (low pressure injection) -LPI flow paths are inoperable; (13) change the surveillance requirement to manually cycle open each LPI-HPI flow path discharge valve every 18 months to require that the HPI discharge crossover valves be cycled every 18 months; and (14) add or modify various administrative and Bases changes that support the proposed changes. The licensee supplied data resulting from risk-informed analyses that were performed in accordance with Regulatory Guides 1.174 and 1.177 to support the evaluation.

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:

No. The proposed change do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. No set points for parameters which initiate protective or mitigative action are being changed.

The proposed changes do not have any impact upon the ability of the HPI [High Pressure Injection] System to add soluble poison to the Reactor Coolant System. The remaining potential impact is upon the ability to mitigate the consequences of a small break LOCA [Loss-of-Coolant Accident], which is addressed below. The small break LOCA is the limiting design basis accident with respect to HPI System operability requirements.

The Technical Specification requirements for the HPI System are supported by a spectrum of small break LOCA analyses based on the approved Evaluation Model described in FTI [Framatome Technologies Incorporated] topical report BAW–10192PA. These small break LOCA analyses demonstrate that the acceptance criteria of 10 CFR 50.46 are satisfied.

The requirements of LCO [Limiting Condition for Operation] 3.5.2 assure that flow can be provided via two HPI trains (i.e., one HPI train responds automatically upon an ESPS [Engineered Safeguards Protective System] signal, and the second HPI train is aligned within 10 minutes via operator actions in the Control Room) following a small break LOCA and a single active failure. The full power small break LOCA analyses supporting this proposed license amendment have been performed in accordance with the approved Evaluation Model described in FTI topical report BAW–10192P.

If enhanced steam generator cooling is not credited in the accident analysis, two HPI trains are required to mitigate specific small break LOCAs with Thermal Power [less than or equal to] 75% RTP [Reactor Thermal Power]. However, if equipment not qualified as QA–1 (i.e., an ADV [Atmosphic Dump Valve] flow path for one steam generator) is credited for enhanced steam generator cooling, the safety analyses have determined that the capacity of one HPI train is sufficient to mitigate a small break LOCA on the discharge of the reactor coolant pumps if Thermal Power [less than or equal to] 75% RTP. An ADV flow path for each steam generator is credited as a compensatory measure in Actions B and C of LCO 3.5.2 to permit operation to continue with THERMAL POWER [less than or equal to] 75% RTP: a) for 30 days with an HPI pump of one or more HPI discharge crossover valve(s) inoperable; and b) for 72 hours with one HPI train inoperable. This provides additional defense-in-depth, because the ADV flow path for each steam generator is required to be operable while only one is needed to perform the function. Additionally, a risk-informed assessment (provided as Attachment 7 to Duke's license amendment request dated December 18, 1998) concluded that operating the plant in accordance with the Required Actions was acceptable.

The proposed changes involve crediting an additional operator action (i.e., steaming that steam generator through an ADV flow path) that has not previously been reviewed and approved by the staff for licensing basis small break LOCA analyses. Additionally, while the EFW System has been credited in past SBLOCA [small break LOCA] analyses as described in responses to NUREG–0565, actions to raise steam generator levels to the loss of subcooled margin setpoint were only assumed in the smaller SBLOCAs. These operator actions have been included in the Emergency Operating Procedure (i.e., AP/1, 2, or 3/A/1800/001) for many years.

The times for completing these operator actions (i.e., feeding a steam generator via EFW [Emergency Feedwater] and steaming that steam generator through an ADV flow path) are new to the small break LOCA analysis and the licensing basis, and are considered reasonable. Crediting the performance of these operator actions within the specified time frames in the SBLOCA analyses does not result in any substantive change to the operator's response to [an] SBLOCA.

In summary, the technical analyses described in this license amendment justify the adequacy of this specification and assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. Therefore, it is concluded that this amendment request will not significantly increase the probability or consequences of an accident previously evaluated.

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

No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. No set points for parameters which initiate protective or mitigative action are being changed. As a result, no new failure modes are being introduced.

The requirements of ITS [Improved Technical Specification] 3.5.2 continue to assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. The requirements are supported by small break LOCA analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are satisfied.

The proposed change involve crediting an additional operator action (i.e., steaming that steam generator through an ADV flow path) that has not previously been reviewed and approved by the staff for licensing basis small break LOCA analyses. Additionally, while the EFW System has been credited in past SBLOCA analyses as described in responses to NUREG–0565, actions to raise steam generator levels to the loss of subcooled margin setpoint were only assumed in the smaller SBLOCAs. These operator actions have been included in the Emergency Operating Procedure (i.e., AP/1, 2, or 3/A/1800/001) for many years.

The times for completing these operator actions (i.e., feeding a steam generator via EFW and steaming that steam generator through an ADV flow path) are new to the small break LOCA analysis and the licensing basis, and are considered reasonable. Crediting the performance of these operator actions within the specified time frames in the SBLOCA analyses does not result in any substantive change to the operator's response to [an] SBLOCA.

Therefore, this proposed amendment will not create the possibility of any new or different kind of accident.

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

No. The requirements of ITS 3.5.2 continue to assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. The requirements are supported by small break LOCA analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are satisfied. These analyses were performed in accordance with the Evaluation Model described in FTI topical report BAW–10192P.

Therefore, it is concluded that the proposed amendment request will not result in a significant decrease in the margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: January 18, 1999

Description of amendment request: The proposed amendments would: (1) delete license condition 2.C.(3) from the Beaver Valley Power Station, Unit No. 1 (BVPS–1) operating license and delete some references to two-loop operation from BVPS–1 Technical Specifications (TSs); (2) revise BVPS–1 and Beaver Valley Power Station, Unit No. 2 (BVPS–2) TS 2.2.1, 3.3.2.1, associated tables 2.2–1 and 3.3.4, and associated bases, to use consistent format and wording between units; (3) revise BVPS–1 and BVPS–2 TS 2.2.1, 3.3.2.1, associated tables 2.2–1 and 3.3.4, and associated bases, to include revised nominal trip setpoints and allowable values which are more conservative than those currently listed; (4) delete or revise TS to reflect the current configuration of Unit 1 plant hardware; and (5) make miscellaneous editorial changes to BVPS–1 and BVPS–2 TS and associated Bases to define terms, revise formatting, modify titles, and add license numbers to pages.

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 [as modified by the NRC staff based upon information provided elsewhere in the licensee's submittal].

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

This proposed amendment includes changes to nominal Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) trip setpoints and allowable values that have been determined with the use of an approved methodology. The new values ensure that all automatic protective actions will be initiated at or before the condition assumed in the safety analysis. This change, which includes modification of the requirements stated in Limiting Safety System Setting (LSSS) 2.2.1 and Limiting Condition for Operation (LCO) 3.3.2.1, will allow the nominal trip setpoints to be adjusted within the calibration tolerance band allowed by the setpoint methodology. There will be no adverse effect on the ability of the channels to perform their safety functions as assumed in the safety analyses. Since there will be no adverse effect on the trip setpoints or the instrumentation associated with the trip setpoints, there will be no significant increase in the probability of any accident previously evaluated.

Other changes in trip system function, content and format are proposed based on the current configuration of the trip system hardware at Beaver Valley Power Station (BVPS) Unit No. 1. Similarly, since the ability of the instrumentation to perform its safety function is not adversely affected, there will be no significant increase in the consequences of any accident previously evaluated.

Since the safety analysis is unaffected by this change there is no change in the consequences of any previously evaluated accident.

The editorial changes do not affect plant safety. The administrative change, for BVPS Unit 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The Technical Specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to [prohibit two-loop operation and] ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valves.

[The administrative change to delete line item 7.d, pertaining to Auxiliary Feedwater (AFW) Pump Auto-start on Emergency Bus Undervoltage, from BVPS–1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic start of the motor-driven AFW pumps is accomplished by the combination of 1) Emergency Bus feed breaker opening 2) valid start signal from ESFAS, and 3) Emergency Diesel Generator (EDG) sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3–3 and 3.3–4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS–1 TS, deleting line item 7.d from BVPS–1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not change the probability or consequences of any accident previously evaluated.]

Therefore, this change does not involve any significant increase in the probability of occurrence 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 amendment includes changes to the format and magnitudes of nominal trip setpoints and allowable values that preserve all safety analysis assumptions related to accident mitigation. The protection system will continue to initiate the protective actions as assumed in the safety analysis. The proposed changes to LSSS 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. This proposed amendment does not involve additional hardware changes. Plant operation will not be changed.

Other proposed changes are made so that the Technical Specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

Furthermore, the proposed changes do not alter the functioning of the RTS and ESFAS. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed RTS and ESFAS trip setpoints are calculated with an approved methodology. The proposed changes to LSSS 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. Therefore, the response of the RTS and ESFAS to accident transients reported in the Updated Final Safety Analysis Report is unaffected by this change. No additional hardware changes are involved. Therefore, accident analysis acceptance criteria are not affected. Other proposed changes are made so that the protection system Technical Specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

The editorial changes do not affect plant safety. The administrative change, for BVPS Unit 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The Technical Specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to [prohibit two-loop operation and] ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valve.

[The administrative change to delete line item 7.d, pertaining to Auxiliary Feedwater (AFW) Pump Auto-start on Emergency Bus Undervoltage, from BVPS–1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic start of the motor-driven AFW pumps is accomplished by the combination of (1) Emergency Bus feed breaker opening, (2) valid start signal from ESFAS, and (3) EDG sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3–3 and 3.3–4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS–1 TS, deleting line item 7.d from BVPS–1 TS Tables 3.3–3, 3.3–4, and 4.3–2 will not involve a significant reduction in a margin of safety.]

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

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

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

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

NRC Project Director: S. Singh Bajwa.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The licensee has proposed an amendment of Facility Operating License No. NPF–47, Appendix A—Technical Specifications, Section 2.1.1.2, entitled ''Reactor Core [Safety Limits].'' The proposed amendment will change the two recirculation loop Minimum Critical Power Ratio (MCPR) limit from 1.13 to 1.12 and the single recirculation loop MCPR limit from 1.14 to 1.13. The revised limits are necessary to address the operation of Cycle 9 following the refueling outage which is scheduled to begin April 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:

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

The plant/cycle specific SLMCPRs have been calculated using methods identical to those used by General Electric (GE) to assess the SLMCPR for other Boiling Water Reactors (BWRs). Similar methods were used to determine the value of the SLMCPR for the previous cycle. These methods are within the existing design and licensing basis and cannot increase the probability or severity of an accident. The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling and fuel damage in the event of the occurrence of Anticipated Operational Occurrences (AOO) or a postulated accident.

The SLMCPR is used to establish the Operating Limit Minimum Critical Power Ratio (OLMCPR). Neither the SLMCPR nor the OLMCPR are initiators or affect initiators of an accident previously evaluated and therefore changes to the SLMCPR do not increase the probability of any accident previously evaluated. The proposed changes involve the use of an accepted methodology in calculating the SLMCPR and, since there is no change in the definition of the SLMCPR, these changes will not affect the consequences of any accident previously evaluated. In addition, the proposed changes do not involve any change in the way the plant is operated. Existing procedures will ensure that the SLMCPR is not violated. Therefore, these changes have no effect on the consequences of an accident.

On these bases, there will be no increase in the probability or consequences of an accident previously analyzed as a result the proposed changes.

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

The proposed changes consist of SLMCPR calculated from an accepted method of analysis that has been used by many BWRs. These changes do not involve any alteration of the plant and do not affect the plant operation. Neither the SLMCPR nor the OLMCPR can initiate an event, therefore a change to the SLMCPR does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

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

The SLMCPR is a Technical Specification numerical value to ensure that 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated. The proposed SLMCPR change results from SLMCPR analysis using the accepted methods as identified in the Attachment.

The margin of safety resides between the SLMCPR and the point at which fuel fails. Maintaining the MCPR above the proposed SLMCPR will maintain the margin of safety associated with GE's SLMCPR methodology. Existing plant procedures will continue to ensure that the SLMCPR is not violated.

Therefore, this request 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: December 23, 1998.

Description of amendment request: The proposed changes will modify the Limiting Condition for Operation for Technical Specifications 3.3.3.7.1 for the chlorine detection system at Waterford Steam Electric Station, Unit 3. A change in the alarm/trip setpoint from 3 parts per million (ppm) to 2 ppm is requested. Additionally, the proposed request corrects a typographical error in Table 3.3–4.

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: The chlorine detection system has no effect on the accidents analyzed in Chapter 15 of the Final Safety Analysis Report. Its only effect is on habitability of the control room, which will be enhanced by specifying a more conservative setpoint in the Technical Specifications (TS). Analysis using more conservative assumptions show that a setpoint of 2 parts per million (ppm) chlorine is acceptable.

Correcting the typographical error on TS page 3/4 3–19 has no effect on the probability or consequences of an accident previously evaluated.

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: The proposed Technical Specification change in itself does not change the design or configuration of the plant. Using a more conservative setpoint performs the same function as the old setpoint, but it accomplishes this function with increased conservatism.

Correcting the typographical error on TS page 3/4 3–19 will not create the possibility of a new or different type of accident from any accident previously evaluated.

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: The chlorine detection system has no effect on a margin of safety as defined by Section 2 of the Technical Specifications. Its only effect is on habitability of the control room, which will be enhanced by a more conservative setpoint provided by this change to the Technical Specifications.

Correcting the typographical error on TS page 3/4 3–19 does not involve a significant reduction in a margin of safety.

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, LA 70122.

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: January 25, 1999.

Description of amendment request: The proposed change request will modify Technical Specification (TS) 3.5.1 to allow up to 72 hours to restore safety injection tank (SIT) to operable status if one SIT is inoperable due to boron concentration not within the limits or the inability to verify level and pressure. The proposed change would also allow up to 24 hours to restore SIT to operable status if one SIT is inoperable due to other reasons when Reactor Coolant System pressure is greater than or equal to 1750 psia. The ACTIONS for an inoperable SIT are being subdivided based on pressurizer pressure to be consistent with the current Waterford 3 requirements and applicability. Additionally, the Surveillance requirement to sample the SIT after a 1% volume increase is being changed to not be required if the source of the makeup is the refueling water storage pool. This amendment request is a collaborative effort of participating Combustion Engineering Owners Group members based on a review of plant operations, deterministic and design basis considerations, and plant risk, as well as previous generic studies and conclusions drawn by the NRC Staff and contained within NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements,'' and NUREG–1432, Revision 1, ''Standard Technical Specifications for Combustion Engineering (CE) Plants.'' TS Bases 3/4.5.1 will be revised to support above changes.

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

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

Response: The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System. The SITs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The SITs were designed to mitigate the consequences of Loss of Coolant Accidents (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analyses. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the affect of the SIT Allowed Outage Time (AOT) extension from 1 hour to 24 hours when one SIT is inoperable for reasons other than boron concentration or inability to measure level or pressure, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD–994, Combustion Engineering Owners Group ''Joint Applications Report for Safety Injection Tank AOT/STI Extension.''

The proposed change to extend the AOT from 1 hour to 72 hours when unable to measure level or pressure is acceptable because SIT operability is not based on instrumentation availability. Therefore, this does not involve a significant increase in the consequences of an accident as evaluated and are endorsed by the Nuclear Regulatory Commission (NRC) in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements.'' The inability to measure level or pressure is acceptable because the SIT instrumentation provides no safety actuation.

The AOT extension from 1 hour to 72 hours, based upon boron concentration outside the prescribed limits does not involve a significant increase in the consequences of an accident as evaluated and approved by the NRC in NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' These changes are acceptable because the reduced concentration effects on core subcriticality during reflood are minor.

The change in sampling requirements to not require sampling if the makeup source is of the same concentration limit as the SIT is acceptable as the concentration will remain within the TS limits.

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

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

Response: The proposed change does not alter the design or configuration of the plant. It also does not alter the mitigation capabilities of any safety system or components. This change increases the AOTs for the condition of SIT inoperability. The boron concentration is maintained by make-up from a source of water with the required concentration of the SITs.

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: The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA and deterministic evaluations were used to evaluate these changes. The PSA evaluations demonstrated that the applicable changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD–994. The deterministic evaluations show that the SITs would be able to perform their safety function. These changes are consistent with NUREG–1366 and NUREG–1432. The margin of safety is not significantly affected by makeup from a source of the same concentration limit as the SIT or increase in the AOT for boron concentration of one SIT not within limits.

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, LA 70122.

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: January 25, 1999.

Description of amendment request: The proposed changes modify Technical Specifications Section 6.0 to remove certain administrative controls and instead rely on the change controls of 10 CFR 50.54(a)(3) and to add a requirement to Section 6.0 concerning the responsibilities of the General Manager Plant Operations. The requested changes are consistent with the Improved Standard Technical Specifications for Combustion Engineering plants, NUREG–1432.

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: The requested changes are purely administrative in nature. The proposed changes do not affect the operation of any structures, systems, or components or the assumptions of any accident analyses. The requested changes only affect Section 6.0 of the Waterford 3 Technical Specifications which describe the administrative controls to be implemented at the site. The requested changes either add an additional administrative requirement or remove quality assurance program details from the Technical Specifications. The details are being removed from the Technical Specifications and instead rely on the change controls of 10 CFR 50.54(a)(3). This submittal makes no changes to the regulatory controls governing changes. The requested changes are purely administrative in nature.

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

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

Response: The proposed changes to the Technical Specification requirements are purely administrative in nature and do not involve a change in plant design or affect the configuration or operation of any structure, system, or component.

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: The proposed changes do not affect the operation of any structures, systems, or components or the assumptions of any accident analyses. The requested changes are purely administrative in nature.

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, LA 70122.

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

NRC Project Director: John N. Hannon.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: January 22, 1999.

Description of amendment request: The proposed amendment would revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Section 4.3, ''Fuel Storage,'' by updating the criticality requirements (k-infinity and U–235 enrichment limits) for storage of fuel assemblies in the spent fuel racks. This change would allow for storage of nuclear fuel assemblies with new designs, including GE–12 with a 10X10 pin array.

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:

After reviewing this proposed amendment, we have concluded:

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

The probability of occurrence of the accident/abnormal conditions evaluated in UFSAR Section 9.1.2.3 is not significantly increased by this change because no modification in fuel handling equipment, fuel pool cooling equipment, fuel storage racks, or fuel handling practices is taking place. Only the k-infinity and enrichment limits for the stored fuel are being changed.

The postulated accident/abnormal conditions evaluated in UFSAR Section 9.1.2.3 have been re-evaluated for the proposed changes in k-infinity and enrichment limits. The results demonstrate that the consequences are negligible. The analyses performed show that the requirement to maintain K-eff less than 0.95 (substantially subcritical) is satisfied for normal and postulated abnormal conditions using methods and assumptions that are consistent with the existing UFSAR. Seismic adequacy and structural integrity of the pool and racks are not affected by the introduction of GE–12 fuel. Local and bulk pool temperatures remain bounded by the current UFSAR analysis for fuel exposures with GE–12 fuel expected through two cycles of operation (i.e., through Cycle 18 operation). Based upon a scoping study comparing the hydraulic diameters of GE–10 and GE–12 fuel, large margins to pool boiling conditions at the final discharge exposures of GE–12 fuel will be maintained. Therefore, the consequences of the accident are not significantly increased by this change.

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

No new types of accidents are being introduced because no modification in fuel handling equipment, fuel pool cooling equipment, fuel storage racks or fuel handling procedures is being made. The design basis function of the spent fuel racks is to maintain the fuel configuration substantially subcritical and within allowable temperatures under both normal and postulated abnormal conditions. This design basis function will be maintained with the proposed k-infinity and enrichment limits.

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

The margin of safety is not significantly reduced. This margin is based on the requirement to limit the K-eff of fuel in the spent fuel racks to less than 0.95. The proposed changes in k-infinity and enrichment limits have been shown to meet this requirement, using methods and assumptions that are consistent with the existing UFSAR. Seismic adequacy and structural integrity of the pool and racks are not affected by the introduction of GE–12 fuel. Local and bulk pool temperatures remain bounded by the current UFSAR analysis for fuel exposures with GE–12 fuel expected through two cycles of operation (i.e., through Cycle 18 operation). Based upon a scoping study comparing the hydraulic diameters of GE–10 and GE–12 fuel, large margins to pool boiling conditions at the final discharge exposures of GE–12 fuel will be maintained.

Based upon the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

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

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401.

Attorney for licensee: Jack Newman, Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Project Director: Cynthia A. Carpenter.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 15, 1998, as supplemented on December 21, 1998.

Description of amendment request: The proposed amendment would revise the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) by adding a new TS 3.7.9, ''Control Building/Standby Gas Treatment System (CB/SBGT) Instrument Air System.'' The proposed amendment would also revise (TS) 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' Condition E, by adding a time limit for plant operation if a penetration flow path is isolated by a single purge valve with resilient seal.

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

The amendment is adding new requirements for the CB/SBGT Instrument Air System that are commensurate with the safety functions it supports and consistent with other support systems in the Technical Specifications. These requirements provide appropriate actions and time limits for plant operation with one or both CB/SBGT Instrument Air subsystems inoperable. The probability of an event while in this condition is low, and the consequences are bounded by the failure of the supported systems. The CB/SBGT Instrument Air System is not assumed to be an initiator of an analyzed event.

The amendment is also adding a time limit for plant operation if a purge valve with resilient seal is used to satisfy TS 3.6.1.3 Required Action E.1 (isolate the affected penetration flow path). While primary containment integrity is provided by the purge valve, it is prudent to limit operation in this condition due to the potential for increased leakage from a single active failure.

These additions will provide assurance that affected systems will be OPERABLE when required and as assumed in the design basis.

This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. This change will not alter assumptions relative to the mitigation of an accident or transient event. This change will not increase the probability of initiating, or the consequences of an analyzed event.

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

The amendment adds new requirements for the CB/SBGT Instrument Air System and adds a time limit for plant operation if a purge valve with resilient seal is used to satisfy TS 3.6.1.3 Required Action E.1.

This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

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

The amendment is adding new requirements for the CB/SBGT Instrument Air System to provide appropriate actions and time limits for plant operation with one or both CB/SBGT Instrument Air subsystems inoperable.

The amendment is also adding a time limit for plant operation if a purge valve with resilient seal is used to satisfy TS 3.6.1.3 Required Action E.1 (isolate the affected penetration flow path). While primary containment integrity is provided by the purge valve, it is prudent to limit operation in this condition due to the potential for increased leakage from a single active failure in the remaining OPERABLE components.

This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. This change will not alter assumptions relative to the primary success path for mitigation of an accident or transient event.

These additions will provide assurance that the accident mitigation functions will perform as assumed in the safety analysis. Thus, the margin of safety will not be reduced.

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: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401.

Attorney for licensee: Jack Newman, Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 29, 1999.

Description of amendment request: The amendment would revise the technical specifications (TS) to relocate three cycle-specific parameter limits; shutdown margin with T_{cold}>210°F, moderator temperature coefficient, and minimum boric acid storage tank level versus concentration, to the Core Operating Limits Report (COLR).

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

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

The safety analysis most impacted by a change to the negative Moderator Temperature Coefficient (MTC) limit is the Main Steam Line Break (MSLB) event. The Steam Line Break Cooldown curves for an MTC are calculated and then input to the cycle-specific MSLB analysis (if necessary) during the reload analysis process, using an NRC-approved methodology. The required/acceptable Shutdown Margin (SDM) is dependent upon the core loading pattern used (i.e., cycle-specific core physics parameters) and is largely dependent on the cycle-specific MTC and available scram worth. The SDM is determined based on the analysis of the Hot Zero Power (HZP) MSLB event in which the return-to-critical and return-to-power conditions are evaluated to provide acceptable results. With the ongoing changes in MTC as a result of core loadings for FCS and higher U–235 enrichments, the end-of-cycle MTC is becoming more negative than the present Technical Specifications limit. Since the MTC is fuel cycle specific and influences the required SDM, it is appropriate to move both of these values to the COLR, consistent with Generic Letter 88–16. Note that no change to the SDM for T_{cold} ≤210°F is being proposed.

The cycle-specific reload analysis is performed for every operating cycle and the results, as incorporated into the COLR pursuant to the 10 CFR 50.59 process, are transmitted to the NRC. FCS will continue to provide COLR updates to the NRC. The relocation of the negative MTC and the ''BAST level versus BAST Concentration'' curves into the COLR, consistent with the NRC recommendations of Generic Letter 88–16, will not modify the methodology used in generating the limits, nor the manner in which they are implemented. These limits will continue to be determined by analyzing the same postulated events as previously analyzed. FCS will continue to operate within the limits specified in the COLR and will take the same corrective actions when or if these limits are exceeded as required by current Technical Specifications. The potential increase of the absolute magnitude of the negative MTC with Shutdown Margin decrease is evaluated during the COLR reload analysis process in accordance with OPPD's NRC-approved topical report. Therefore, this proposed amendment is administrative in nature and has been concluded not to increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to FCS Technical Specifications were the result of a recommendation from a Generic Letter. Future changes to the parameters being relocated to the COLR can only be performed with approved Reload Analyses. No new or different kind of accident is created by this administrative change because the actual operation of FCS remains unchanged. Therefore the possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report would not be created.

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

As indicated above, the implementation of this proposed COLR change, consistent with the guidance of Generic Letter 88–16, makes use of the existing safety analysis methodologies and the resulting limits and setpoints for plant operation. Additionally, the safety analysis acceptance criteria for operation with this proposed amendment have not changed from the criteria used in the current reload analysis. Therefore, the margin of safety as defined in the bases of Technical Specifications is not reduced.

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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: January 12, 1999.

Description of amendment request: The proposed change involves revising Technical Specification (TS) Section 3/4.4.2, ''Safety/Relief Valves,'' and TS Bases Sections B 3/4.4.2, B 3/4.5.1 and B 3/4.5.2, to increase the allowable as-found main steam Safety Relief Valve (SRV) code safety function lift setpoint tolerance from plus or minus 1% to plus or minus 3%. This change will also require the as-left SRV code safety function lift setting to be set within plus or minus 1% of the specified nominal lift setpoint prior to reinstallation in the plant. In support of this proposed TS change, the required number of OPERABLE SRVs in Operational Conditions (OPCONs) 1, 2, and 3 will be changed from 11 to 12. The number of SRVs in each lift pressure grouping will remain the same. This proposed TS change does not alter the SRV nominal lift setpoints or the SRV lift setpoint test frequency currently specified by TS Section 3/4.4.2. The proposed change does not change the SRV testing commitment specified in LGS Updated Final Safety Analysis Report (UFSAR) Chapter 5.2.2.10, ''Inspection and Testing.''

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 Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes allow for an increase in the as-found main steam Safety Relief Valve (SRV) setpoint tolerance from plus or minus 1% to plus or minus 3%. The proposed changes also reduce the allowable number of SRVs to be out-of-service from three (3) to two (2). The proposed changes do not alter the SRV nominal lift setpoints or SRV lift setpoint test frequency. The actuation of an SRV is the precursor to the inadvertent opening of a SRV transient, as discussed in Updated Final Safety Analysis Report (UFSAR) Chapter 15.1.4. Increasing the allowable as-found SRV code safety function lift setpoint tolerance from plus or minus 1% to plus or minus 3% does have the potential for the minimum SRV simmer margin to be reduced from 113.3 psig to 89.9 psig. A reduction in simmer margin will not directly result in an increase of the probability on an inadvertent self actuation of an SRV. A reduction in simmer margin will reduce the seating force which may initiate leakage. However, this leakage is monitored and corrective actions can be implemented prior to progressing to the point of the potential of an inadvertent actuation. This reduction in SRV simmer margin has been evaluated by the SRV manufacturer and determined to be acceptable; therefore, the probability of an inadvertent SRV actuation remains unchanged. Actuation of an SRV is not a precursor for any other event evaluated in the Safety Analysis Report (SAR).

The proposed TS changes have been evaluated on both a generic and plant specific basis. The NRC has approved the general approach of this change; however, implementation is contingent on several plant specific evaluations. The required plant specific analyses and evaluations included transient analysis of the anticipated operational transients (AOTs); analysis of the design basis overpressurization event; evaluation of the performance of high pressure systems, motor operated valves, and vessel instrumentation and associated piping; and evaluation of the containment response during Loss-of-Coolant Accident (LOCA) and hydrodynamic loads on the SRV discharge lines and containment. In addition to the plant specific analyses and evaluations required by the NRC, the following items were also considered: ECCS/LOCA [Emergency Core Cooling System] performance, SRV simmer margin, high pressure—low pressure interfaces, i.e., High Energy Line Break (HELB), Station Blackout (SBO), and Fire Safe Shutdown (FSSD), and the short term pressurization phase of an ATWS [anticipated transient without scram] event. These analyses and evaluations show that there is adequate margin to the design core thermal limits and reactor vessel pressure limits using the plus or minus 3% SRV code safety function lift setpoint tolerance and two (2) SRVs out-of-service. The analyses and evaluations also show that the operation of the high pressure injection systems will not be adversely affected, that SRV discharge piping stresses will not be exceeded, and that the containment response during a LOCA will be acceptable.

Evaluations of the impact of the proposed change on the Equipment Important to Safety have been performed and no adverse conditions were identified. The reactor pressure vessel and attached systems and piping have been evaluated for the impact of this proposed TS change. A plant specific analysis has been performed which indicates that neither the American Society of Mechanical Engineers (ASME) Code upset limits or the TS Safety Limits for the reactor pressure vessel will be exceeded for the limiting event, i.e., Main Steam Isolation Valve (MSIV) closure with flux Scram. The reactor pressure vessel and attached piping design values will not be exceeded. The current high pressure—low pressure interface evaluation utilized nominal SRV setpoints, and therefore, is unaffected. Therefore, the probability of a malfunction of the reactor pressure vessel and attached systems and piping is not increased.

The nuclear fuel has been evaluated for the impact of the proposed change. Plant specific analyses were performed which indicate that for all abnormal operational transients adequate margin to the limiting thermal limit parameter, i.e., Minimum Critical Power Ratio (MCPR), is maintained. Emergency Core Cooling System (ECCS)/LOCA performance is maintained adequate to meet the requirements of 10CFR50.46. Therefore, the probability of the malfunction of the nuclear fuel is not increased.

The SRVs have been evaluated for the impact of the proposed TS changes. No physical changes to the SRVs will be made as a result of the proposed TS changes. Adequate simmer margin will be maintained with the increased tolerance to ensure that an inadvertent lifting of a SRV does not occur.

The increase in SRV discharge flow and reactor vessel pressure due to the potential for higher SRV lift setpoints are bounded by the SRV steam flows and reactor vessel pressure currently used in the evaluation of SRV discharge piping, quencher, quencher support, and hydrodynamic loads on the suppression pool and submerged structures; therefore, the probability of a malfunction of a SRV or associated components and structures is not increased.

The Containment response during a LOCA has been evaluated for the impact of the proposed change. The major factor in the Containment response to a LOCA is the rate of reactor vessel water inventory loss. The rate of reactor vessel water inventory loss is mainly dependent on reactor decay heat which is not affected by the proposed change. Therefore, the probability of the malfunction of the Containment is not increased.

The High Pressure Coolant Injection (HPCI) system has been evaluated for the impact of the proposed TS changes. The analysis determined that the HPCI system would not be capable of developing its design flowrate of 5600 gpm at a reactor pressure of 1205 psig (lowest SRV nominal setpoint +3% tolerance) unless the HPCI turbine/pump maximum rated speed was increased. However, increasing the HPCI turbine/pump maximum rated speed is prevented due to HPCI pump discharge piping overpressurization concerns. Further analysis has shown that the HPCI system is capable of meeting its required ECCS function design flowrate, and its required non-ECCS flowrate, without any change to the current system operating parameters. Therefore, the probability of a malfunction of the HPCI System is not increased.

The Reactor Core Isolation Cooling (RCIC) system has been evaluated for the impact of the proposed change. The analysis determined that in order for the RCIC system to be capable of injecting its design flowrate of 600 gpm at a reactor pressure of 1205 psig (lowest SRV setpoint of 1170 psig +3% tolerance) the maximum rated speed of the RCIC turbine/pump is required to be increased from 4575 rpm to 4625 rpm. This increase in the RCIC turbine/pump maximum rated speed will reduce the margin to the overspeed trip from 123% to 122.1%. This reduction in the margin to the overspeed trip is acceptable due to the implementation of plant Modification P00210, ''RCIC System Startup Transient Improvement,'' which reduced the amount of turbine/pump speed overshoot during system startup. The RCIC overspeed trip setpoint will not be changed; therefore, a failure of the RCIC turbine/pump (missile hazard or system overpressurization) due to overspeed is not increased. All other RCIC System components will continue to operate within the currently specified design and operating limits. Therefore, the probability of a malfunction of the RCIC System is not increased.

The Standby Liquid Control (SLC) system has been evaluated for the impact of the proposed change. The SLC system capability of shutting down the reactor during a postulated event in which all or some of the control rods cannot be inserted or during a postulated Anticipated Transient Without Scram (ATWS) event is not impacted by this proposed change. Therefore, the probability of a malfunction of the SLCS is not increased.

The Control Rod Drive (CRD) system has been evaluated for the impact of the proposed change. The CRD system capability of controlling reactor power during normal plant operation and rapidly inserting control rod blades (Scram) during abnormal plant conditions is not impacted by the proposed change. Therefore, the probability of a malfunction of the CRD system is not increased.

The Reactor Vessel Instrumentation System has been evaluated for the impact of the proposed change. The Reactor Vessel Instrumentation System will continue to be operated within the current design pressure/temperature requirements; therefore, the probability of a malfunction of the Reactor Vessel Instrumentation System is not increased.

The LGS, Units 1 and 2, Generic Letter 89–10 Motor-Operated Valve (MOV) Program has been evaluated for the proposed change. The LGS MOV Program currently uses SRV nominal setpoints for differential pressure determinations for valves in which reactor pressure at the SRV setpoint is limiting. Use of nominal SRV setpoints is consistent with current industry practice. Therefore, the probability of a malfunction of a MOV is not increased.

Reducing the number of SRVs allowed to be out-of-service does not make the consequences of a malfunction of a SRV more severe, since the number of SRVs required to maintain the reactor vessel within ASME Code and TS Safety Limits will be maintained OPERABLE. The proposed change does not result in any changes to the interactions of any system, structure, or component. All systems, structures, and components will continue to function as designed.

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

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

The proposed TS changes allow for an increase in the as-found SRV setpoint tolerance from plus or minus 1% to plus or minus 3%. The proposed TS changes also reduce the allowable number of SRVs to be out-of-service from three (3) to two (2). Generic and plant specific analyses and evaluations indicate that the plant response to any previously evaluated event will remain unchanged. All plant systems, structures, and components will continue to be capable of performing their required safety function as required by event analysis guidance.

The proposed TS changes do not alter the SRV nominal lift setpoints or SRV lift setpoint test frequency. The operation and response of the affected Equipment Important to Safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. No system, structure, or component will be subjected to a condition that has not been evaluated and determined to be acceptable using the guidance required for specific event analysis.

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

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

The proposed TS changes allow for an increase in the as-found SRV setpoint tolerance from plus or minus 1% to plus or minus 3%. The proposed TS changes also reduce the allowable number of SRVs to be out-of-service from three (3) to two (2). The proposed TS changes do not alter the SRV nominal lift setpoints or SRV lift setpoint test frequency. The operation and response of the affected Equipment Important to Safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. While the calculated peak reactor vessel pressure for the ASME overpressure event and the ATWS Pressure Regulator Failure-Open (PREGO) event are higher than those calculated without the increase in setpoint tolerance, both are still within the respective licensing acceptance limits associated with these events. These licensing acceptance limits have been determined by the NRC to provide a sufficient margin of safety.

The increase in the RCIC system turbine/pump maximum rated speed is within the capability of the system design. The reduction in the margin to the overspeed trip is not a reduction in the margin of safety, since the operation of the RCIC System has demonstrated minimal speed overshoot on system initiation due to the installation of plant Modification P00210, ''RCIC System Startup Transient Improvement.''

The inability of the HPCI system to be capable of injecting 5600 gpm at a reactor pressure of 1205 psig (lowest SRV nominal setpoint of 1170 psig +3% tolerance) is not a reduction in the margin of safety, since analysis for events that would result in high reactor vessel pressure indicate that the HPCI System is capable of providing adequate coolant injection.

The increase in SRV steam flow and reactor vessel pressure does not reduce the margin of safety associated with the SRVs and associated components and structures since the increased SRV steam flow rate and reactor vessel pressure are bounded by the current design analysis.

The margin of safety for fuel thermal limits and 10CFR50.46 limits is unaffected by the proposed change.

The margin of safety for the Containment is unaffected by the proposed change.

The capability of the SLC system to perform its safety function during all required events, using the required guidance for event analysis, is maintained. Therefore, the proposed changes do not reduce the margin of safety provided by the SLC system.

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

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: January 25, 1999.

Description of amendment request: The proposed Technical Specification (TS) Change Request revises the TS Surveillance Requirement frequencies for Sections 4.8.1.1.2.e.1, 4.8.1.1.2.e.8.a, and 4.8.1.1.2.e.8.b for the Emergency Diesel Generator maintenance inspection outages, the 24-hour endurance run, and for the hot restart test from 18 to 24 months.

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 Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The maintenance inspection interval change and the corresponding interval change for the associated 24 hour endurance test and hot restart test which are normally performed in conjunction with the diesel preventive maintenance overhaul inspections, as well as the programmatic improvements addressed here do not involve physical changes that would affect the ability of the EDGs [emergency diesel generators] to perform their safety function. The Emergency Diesel Generator System is not an accident initiator.

The Surveillance Testing requirements of Technical Specification Section 3/4.8 will continue to verify the operability and reliability of the Emergency Diesel Generator system.

The proposed changes do not affect the ability of the EDGs to mitigate the consequences of an accident, including the Loss of Coolant Accident (LOCA) coupled with Loss Of Offsite Power accident analyses as presented in Chapter 15 of the LGS [Limerick Generating Station] UFSAR [Updated Final Safety Analysis Report]. EDG unavailability due mostly to outage inspections is more than 2 times higher than EDG unplanned unavailability. An extension of the outage inspection frequency to 24 months will result in increased EDG availability to mitigate the consequences of a potential accident. When this program is taken in its entirety the extended maintenance intervals coupled with the defined enhancements is judged to result in an overall increase in EDG availability and reliability. Therefore, the probability or consequences of an accident previously evaluated is not increased.

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

The Emergency Diesel Generator system is not an accident initiator. The operation and design of the onsite emergency power system (including the EDGs) is not being changed; only the overhaul inspection interval coupled with the program improvements and the corresponding interval change for the associated 24 hour endurance test and hot restart test, (which are normally performed in conjunction with the diesel preventive maintenance overhaul inspections), are changed. The EDG system meets the single failure criteria at the EDG unit level, i.e., the SAR [safety analysis report] states that with one EDG failed or out-of-service, the standby AC system is capable of furnishing sufficient power for the minimum Class 1E load demand, assuming a limiting design basis accident has occurred. The proposed changes involve a routine preventive maintenance and inspection time interval change along with the corresponding surveillance test interval changes, and also include programmatic improvements to reduce the likelihood of a failure of an individual EDG unit; the proposed changes do not involve any physical design or operational changes that could create a malfunction extending beyond an individual EDG nor do they increase the potential for a common-mode EDG failure. Therefore, it is not possible to create a new or different type of accident through implementation of these changes.

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

The changes to bring the frequencies of the EDG overhaul, the 24 hour endurance test and the associated hot restart test into alignment with the current 2 year operating cycle, and the detailed programmatic changes to achieve conformance with the FMOG [Fairbanks Morse Owners Group] recommended maintenance program, will increase the reliability and availability of the EDG system. This will enhance the margin of safety as the amount of time the EDGs are out-of-service will decrease and the system will be single-failure proof for more clock hours when the nuclear reactor(s) are operating. The changes discussed here do not result in operation of the emergency diesel generator system nor any other plant system in a manner beyond their original design basis, and thus does not reduce any explicit or implicit Technical Specification 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: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

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

NRC Project Director: William M. Dean.

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

Date of application for amendment: February 12, 1997.

Brief description of amendment: The proposed amendment would delete a portion of the Trojan site from the 10 CFR 50 license when that portion of the site, designated for use as an independently licensed spent fuel storage installation (ISFSI), receives a part 72 license.

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. The NRC staff has reviewed the licensees' analysis against the standards of 10 CFR 50.92(c). The licensee's analysis is summarized below:

The proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change is administrative in nature and has no impact on the probability or consequences of accidents previously evaluated. The physical structures, systems, and components of the Trojan Nuclear Plant and the operating procedures for their use are unaffected by this proposed change. The proposed action would eliminate the ISFSI area from the Part 50 license when the Part 72 license is issued. The 10 CFR 72 licensing controls for the area will assure an adequate level of safety for the area during normal operation of the ISFSI and during abnormal events or accidents. Therefore the proposed Part 50 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed action would eliminate the ISFSI area from the Part 50 license when the Part 72 license is issued. The proposed change is administrative in nature and has no impact on plant systems, structures, or components or on any procedures for operating the plant equipment. The ISFSI will be separately licensed under Part 72 and physically separated from the Part 50 licensed structures and equipment. Therefore, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

The proposed changes do not involve reduction in the margin of safety. The Trojan Permanently Defueled Technical Specifications (PDTS) contain four limiting conditions of operation that address: 1) Spent Fuel Water Level, 2) Spent Fuel Pool Boron Concentration, 3) Spent Fuel Pool Temperature, and 4) Spent Fuel Pool load restrictions. These PDTS will remain in effect as long as spent fuel is stored in the Spent Fuel Pool, which is in accordance with their applicability statements. The ISFSI area is physically separated from the Spent Fuel Pool area and the Fuel Building and will have no effect on spent fuel water level, spent fuel pool boron concentration, spent fuel pool temperature, or loads over the Spent Fuel Pool. The proposed change is administrative and does not affect plant equipment, operating parameters, or procedures. Based on the above, the proposed change will not reduce the margin of safety.

Based on a staff review of the licensee's analysis, 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: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

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

NRR Project Director: Seymour H. Weiss.

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

Date of application for amendment: January 7, 1999.

Brief description of amendment: The proposed amendment would allow loading and handling of spent fuel transfer and storage casks in the Trojan Fuel Building.

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. The NRC staff has reviewed the licensees' analysis against the standards of 10 CFR 50.92(c). The licensee's analysis is summarized below:

The proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated. With the permanent cessation of operations, the number of potential accidents was reduced to those types of accidents associated with the storage of irradiated fuel and radioactive waste storage and handling. Additional events were postulated for decommissioning activities due to the difference in the types of activities that were to be performed. The postulated accidents in the Defueled Safety Analysis Report (DSAR) are generally classified as: (1) radioactive release from a subsystem or component, (2) fuel handling accident and, (3) loss of spent fuel decay heat removal capability. The postulated events described in the Decommissioning Plan are grouped as: (1) decontamination, dismantlement, and materials handling events, (2) loss of support systems (offsite power, cooling water, and compressed air), (3) fire and explosions, and (4) external events (earthquake, external flooding, tornadoes, extreme winds, volcanoes, lightning, toxic chemical release). These types of accidents are discussed below.

Radioactive release from a subsystem or component involves failure of a radioactive waste gas decay tank (WGDT) or failure of a chemical and volume control system holdup tank (HUT). For a failure of a WGDT, the radioactive contents are assumed to be principally the noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes and trace quantities of halogens. For the failure of a HUT, the assumptions were full power operations with 1-percent failed fuel, 40 weeks elapsed since power operation, and 60,000 gallons of 120° F liquid released over a 2-hour period. However, the WGDT's and HUT's are no longer active and have been emptied. Therefore, cask loading and transfer activities cannot increase the probability of occurrence of a failure or the consequence of a failure of the WGDT's or HUT's.

The fuel handling accident involves a stuck or dropped fuel assembly that results in damage of the cladding of the fuel rods in one assembly and the release of gaseous fission products. Spent fuel handling and loading will involve moving the spent fuel assemblies one by one, from the Spent Fuel Pool to the baskets which will be located in the Cask Loading Pit. The fuel handling equipment will be the same as had been previously analyzed with the exception of special tools which will be used to manipulate failed fuel. These special tools will be similar in size and weight to the existing tools used for underwater manipulation and therefore will not present a new hazard. In addition, the same administrative controls and physical limitations imposed on any fuel handling operation will be used for spent fuel loading and handling. The potential release, 100 percent of gap noble gas, from a fuel assembly is not affected (although the fission product inventory in a fuel assembly continues to decrease with time). Thus there is no increase in the probability of occurrence or consequences of a fuel handling accident over what would be expected for any routine fuel handling operation.

The loss of spent fuel decay heat removal capability involves the loss of forced spent fuel cooling with and without concurrent Spent Fuel Pool inventory loss. The only requirement to assure adequate decay heat removal capability for the spent fuel is to maintain the water level in the Spent Fuel Pool so that the fuel assemblies remain covered (i.e. the capability to make up water to the Spent Fuel Pool must be available when required). The potential events which could result in a loss of spent fuel decay heat removal include external events (explosions, toxic chemical, fires, ship collision with intake structure, oil or corrosive liquid spills in the river, cooling tower collapse, seismic events, severe meteorological events), and internal events including Spent Fuel Pool makeup water system malfunctions (Service Water System, electrical power, instrument air). Spent fuel loading and handling will not require the use of explosive materials (the gases used for electric arc welding are inert), toxic chemicals or flammable materials (routine use of contamination control materials is not considered to present a significant hazard). The probability of other external events (e.g. cooling tower collapse) is not effected by the spent fuel handling and loading activities inside the Fuel Building. Spent fuel loading and handling activities will not directly interface with the Spent Fuel Pool makeup water systems, therefore does not affect their probability of failure. (The Cask Loading Pit will be filled with borated water from the Spent Fuel Pool that will be cooled by the Spent Fuel Cooling System, but use of this water in the Cask Loading Pit does not increase the failure probability of the Spent Fuel Pool or makeup water systems.) As described in the licensees' safety evaluation, the safe load path and handling height limitations will ensure that a load drop does not adversely affect the Spent Fuel Pool or the makeup water systems. Therefore there is no significant increase in the probability or consequences of a loss of spent fuel decay heat removal capability.

The events postulated in the Decommissioning Plan are similar to the DSAR with the exception of the decontamination, dismantlement, and materials handling events. Decontamination events involve gross liquid leakage from in-situ decontamination equipment (e.g. tanks) or accidental spraying of liquids containing concentrated contamination. Dismantlement events involve segmentation of components and structures, or removal of concrete by rock splitting, explosives, or electric and/or pneumatic hammers. Dismantlement events potentially result in airborne contamination. Material handling events involve the dropping of contaminated components, concrete rubble, filters, or packages of particulate materials. Licensee administrative controls will be implemented to ensure that spent fuel loading and handling activities and decommissioning activities will not be performed concurrently if they interact with each other and could increase the probability or consequences of a postulated event of accident. Therefore, neither the probability nor the consequences of decontamination, dismantlement, and materials handling events will not be significantly increased.

The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. As described in the licensees' safety evaluation the potential accidents associated with fuel handling and loading were similar to fuel handling accidents, material handling events and pressurized line break previously analyzed. Additionally the potential consequences were a small fraction of Environmental Protection Agency (EPA) Protective Action Guides (PAG's). Therefore, fuel loading and handling does not present new or different types of accidents.

The proposed changes do not involve a significant reduction in the margin of safety. The Trojan Permanently Defueled Technical Specifications (PDTS) contain four limiting conditions of operation that address: (1) Spent fuel water level, (2) spent fuel pool boron concentration, (3) spent fuel pool temperature, and (4) spent fuel pool load restrictions. These PDTS will remain in effect as long as spent fuel is stored in the Spent Fuel Pool, which is in accordance with their applicability statements. The spent fuel loading and handling activities will not affect these PDTS or their bases.

The Cask Loading Pit, where the spent fuel will be loaded into the basket, is immediately adjacent to the Spent Fuel Pool. The gate between the Cask Loading Pit and Spent Fuel Pool will be open to allow transfer of spent fuel assemblies from storage racks in the Spent Fuel Pool to the basket in the Cask Loading Pit. Opening the gate between them will allow free exchange of water between the Cask Loading Pit and the Spent Fuel Pool. The Cask Loading Pit will be filled with borated water at approximately the same concentration and temperature as the Spent Fuel Pool prior to opening the gate. This will maintain the limiting conditions for operation for Spent Fuel Pool boron concentration, temperature, and water level and the margin of safety will not be affected.

Spent fuel loading and handling activities will involve lifting and moving heavy loads (e.g. transfer cask, basket). Loads that will be carried over fuel in the Spent Fuel Pool racks and the heights at which they will be carried will be limited to preclude impact energies over 240,000 in-lbs if the loads were dropped. This is in accordance with limiting condition for operation 3.1.4 ''Spent Fuel Pool Load Restrictions.'' With this precaution, the limiting condition for operation pertaining to load restrictions over the Spent Fuel Pool will be satisfied and the margin of safety will be unaffected. The safe load paths for heavy loads being lifted outside the Spent Fuel Pool will be sufficiently far from the Spent Fuel Pool so as to not have an interaction in the unlikely event of a load drop. In addition mechanical stops and electrical interlocks on the Fuel Building overhead crane will provide additional assurance that heavy loads are not carried over the Spent Fuel Pool racks.

Based on the above, the spent fuel loading and handling activities will not reduce the margin of safety.

Based on a staff review of the licensee's analysis, 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: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

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

NRR Project Director: Seymour H. Weiss.

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

Date of application for amendment: January 27, 1999.

Brief description of amendment: The proposed amendment would allow unloading of spent fuel transfer casks in the Trojan Fuel Building.

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The licensee's analysis is summarized below:

The proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated. With the permanent cessation of operations, the number of potential accidents was reduced to those types of accidents associated with the storage of irradiated fuel and radioactive waste storage and handling. Additional events were postulated for decommissioning activities due to the difference in the types of activities that were to be performed. The postulated accidents in the Defueled Safety Analysis Report (DSAR) are generally classified as: (1) Radioactive release from a subsystem or component, (2) fuel handling accident and, (3) loss of spent fuel decay heat removal capability. The postulated events described in the Decommissioning Plan are grouped as: (1) Decontamination, dismantlement, and materials handling events, (2) loss of support systems (offsite power, cooling water, and compressed air), (3) fire and explosions, and (4) external events (earthquake, external flooding, tornadoes, extreme winds, volcanoes, lightning, and toxic chemical release). These types of accidents are discussed below.

Radioactive release from a subsystem or component involves failure of a radioactive waste gas decay tank (WGDT) or failure of a chemical and volume control system holdup tank (HUT). For a failure of a WGDT, the radioactive contents are assumed to be principally the noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes and trace quantities of halogens. For the failure of a HUT, the assumptions were full power operations with 1-percent failed fuel, 40 weeks elapsed since power operation, and 60,000 gallons of 120° F liquid released over a two hour period. However, the WGDT's and HUT's are no longer active and have been emptied. Therefore, cask loading and transfer activities cannot increase the probability of occurrence of a failure or the consequence of a failure of the WGDT's or HUT's.

The fuel handling accident involves a stuck or dropped fuel assembly that results in damage of the cladding of the fuel rods in one assembly and the release of gaseous fission products. Spent fuel cask unloading will involve moving the spent fuel assemblies one by one, from the baskets which will be located in the cask loading pit to the spent fuel pool. The fuel handling equipment will be the same as had been previously analyzed. In addition, the same administrative controls on physical limitations imposed on fuel handling and fuel loading operations will be used for fuel unloading. The potential release, 100 percent of noble gases within the gap, from a fuel assembly is not affected (although the inventory in a radioactive stored fuel assembly continues to decrease with time). Thus, there is no increase in the probability of occurrence or consequences of a fuel handling accident over what would be expected for any routine fuel handling operation or loading of fuel into a cask.

The loss of spent fuel decay heat removal capability involves the loss of forced spent fuel cooling with and without concurrent spent fuel pool inventory loss. The only requirement to assure adequate decay heat removal capability for the spent fuel is to maintain the water level in the spent fuel pool so that the fuel assemblies remain covered (i.e., the capability to make up water to the spent fuel pool must be available when required). The potential events that could result in a loss of spent fuel decay heat removal include external events (explosions, toxic chemical, fires, ship collision with intake structure, oil or corrosive liquid spills in the river, cooling tower collapse, seismic events, and severe meteorological events), and internal events including spent fuel pool makeup water system malfunctions (service water system, electrical power, and instrument air). Spent fuel cask unloading will not require the use of explosive materials, toxic chemicals or flammable materials (routine use of contamination control materials is not considered to present a significant hazard). The probability of other external events (e.g. cooling tower collapse) is not effected by the spent fuel unloading activities inside the fuel building. Spent fuel cask unloading activities will not directly interface with the spent fuel pool makeup water systems, and therefore does not affect their probability of failure. (The cask loading pit will be filled with borated water from the spent fuel pool that will be cooled by the spent fuel cooling system, but use of this water in the cask loading pit does not increase the failure probability of the spent fuel pool or makeup water systems). As described in the licensees' safety evaluation, the safe load path and handling height limitations will ensure that a load drop does not adversely affect the spent fuel pool or the makeup water systems. Therefore, there is no significant increase in the probability or consequences of a loss of spent fuel decay heat removal capability.

The events postulated in the Decommissioning Plan are similar to the DSAR with the exception of the decontamination, dismantlement, and materials handling events. Decontamination events involve gross liquid leakage from in-situ decontamination equipment (e.g. tanks) or accidental spraying of liquids containing concentrated contamination. Dismantlement events involve segmentation of components and structures, or removal of concrete by rock splitting, explosives, or electric and/or pneumatic hammers. Dismantlement events potentially result in airborne contamination. Material handling events involve the dropping of contaminated components, concrete rubble, filters, or packages of particulate materials. Licensee administrative controls will be implemented to ensure that spent fuel cask unloading activities and decommissioning activities will not be performed concurrently if they interact with each other and could increase the probability or consequences of a postulated event of accident. Therefore, neither the probability nor the consequences of decontamination, dismantlement, and materials handling events will be significantly increased.

The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. As described in the licensee's safety evaluation the potential accidents associated with fuel cask unloading were similar to fuel handling accidents, material handling events and pressurized line break previously analyzed. Additionally the potential consequences were a small fraction of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). Therefore, fuel loading and handling does not present new or different types of accidents.

The proposed changes do not involve a significant reduction in the margin of safety. The Trojan Permanently Defueled Technical Specifications (PDTS) contain four limiting conditions of operation that address: (1) spent fuel pool water level, (2) spent fuel pool boron concentration, (3) spent fuel pool temperature, and (4) spent fuel pool load restrictions. These PDTS will remain in effect as long as spent fuel is stored in the spent fuel pool, which is in accordance with their applicability statements. The spent fuel cask unloading activities will not affect these PDTS or their bases.

The cask loading pit, where the spent fuel will be unloaded from basket, is immediately adjacent to the spent fuel pool. The gate between the cask loading pit and spent fuel pool will be open to allow transfer of spent fuel assemblies from the basket in the cask loading pit to the storage racks in the spent fuel pool. Opening the gate between them will allow free exchange of water between the cask loading pit and the spent fuel pool. The cask loading pit will be filled with borated water at approximately the same concentration and temperature as the spent fuel pool prior to initial cask loading. This will maintain the limiting conditions for operation for spent fuel pool boron concentration, temperature, and water level and the margin of safety will not be affected.

Spent fuel cask unloading activities may involve lifting and moving heavy loads (e.g. transfer cask, basket). Loads that will be carried over fuel in the spent fuel pool racks and the heights at which they will be carried will be limited to preclude impact energies over 240,000 in-lbs if the loads were dropped. This is in accordance with limiting condition for operation 3.1.4 ''Spent Fuel Pool Load Restrictions.'' With this precaution, the limiting condition for operation pertaining to load restrictions over the spent fuel pool will be satisfied and the margin of safety will be unaffected. The safe load paths for heavy loads being lifted outside the spent fuel pool will be sufficiently far from the spent fuel pool so as to not have an interaction in the unlikely event of a load drop. In addition, mechanical stops and electrical interlocks on the fuel building overhead crane will provide additional assurance that heavy loads are not carried over the spent fuel pool racks.

Based on the above, the spent fuel cask unloading activities will not reduce the margin of safety.

Based on a staff review of the licensee's analysis, 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: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

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

NRR Project Director: Seymour H. Weiss.

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: October 16, 1998, as supplemented January 28, 1999.

Description of amendment request: This application for amendment to the Indian Point 3 (IP3) Technical Specifications (TSs) proposes to relocate the Chemical Volume and Control System (CVCS) TS 3.2 from the TSs to the IP3 Operational Specifications.

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

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

Response: Relocation (i.e., removal from TS) of TS 3.2, the bases and the associated surveillances in Table 4.1–1 (items 12, 26, and 27), Table 4.1–2 (item 2), and Table 4.1–3 (item 12) will not involve a significant increase [in] the probability or consequences of an accident since the relocation of the Technical Specifications to administrative controls governed by 10 CFR 50.59 does not affect the availability or function of charging and boric acid flow paths. CVCS is not an initiator of an accident (the dilution event is equipment malfunction that is manually terminated) and the proposed change does not alter overall system operation, physical design, system configuration, or operational setpoints. There will be no significant increase in the consequences of an accident because the required boration flow paths will continue to be available for boration to the reactor coolant system.

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

Response: Relocation (i.e., removal from TS) of TS 3.2, the bases and the associated surveillances in Table 4.1–1 (items 12, 26, and 27), Table 4.1–2 (item 2), and Table 4.1–3 (item 12) will not create the possibility of a new or different kind of accident from any previously evaluated since it does not alter the overall system operation, physical design, system configuration, or operational setpoints. The plant systems for boration are operated in the same manner as before and, consequently, the relocation does not introduce any new accident initiators or failure mechanisms and does not invalidate the existing dilution event response. The boration function is not an accident initiator.

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

Response: Relocation (i.e., removal from TS) of TS 3.2, the bases and the associated surveillances in Table 4.1–1 (items 12, 26, and 27), Table 4.1–2 (item 2), and Table 4.1–3 (item 12) will not involve a significant reduction in margin of safety. The relocation is a change to the administrative controls that are used to assure system availability and those administrative controls are governed by 10 CFR 50.59. The manner in which the system is operated does not change and there is no change to physical design, system configuration, or operational setpoints. Previous analyses of system malfunction remain unchanged. The current Technical Specification does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for inclusion in the technical specifications.

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 Project Director: S. Singh Bajwa, Director.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: December 30, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.3 and Table 3.7.3–1. The proposed changes would modify the flood protection actions required during periods of elevated river water level.

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

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

The proposed TS revisions related to flood protection TS Action Statements involve no hardware changes and no changes to existing structures, systems or components. The proposed changes to the flood protection TS Action Statements ensure that the supported systems can perform their required safety functions under worst case design basis conditions, consistent with limitations imposed by other TS. The proposed flood protection TS ACTION Statements ensure that the plant is directed to enter a safe shutdown condition whenever the capability to withstand worst case design basis conditions is affected. Since the flood protection changes will still ensure that the plant remains capable of meeting applicable design basis requirements and retains the capability to mitigate the consequences of accidents described in the [Hope Creek] HC [Updated Final Safety Analysis Report] UFSAR, the proposed changes were determined to be acceptable. As a result, these changes will neither increase the probability of an accident previously evaluated nor increase the radiological dose consequences of an accident previously evaluated.

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

The proposed changes to the flood protection TS contained in this submittal will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve no hardware changes and no changes to existing structures, systems or components, there can be no impact on the potential occurrence of any accident due to new equipment failure modes. The resulting operational limits imposed by the flood protection LCO ensure that the plant can either perform its design basis safety functions or an appropriately conservative shutdown action statement is entered. Furthermore, there is no change in plant testing proposed in this change request that could initiate an event. Therefore, these changes 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 changes for the flood protection TS retain the plant's continued capability to withstand worst case design basis conditions. The proposed flood protection TS ACTION Statements ensure that the plant is directed to: (1) enter a safe shutdown condition whenever the capability to withstand worst case design basis conditions is lost; or (2) enter a conservatively short period of continued operation when supported system redundancy is reduced. Since the plant will still remain capable of meeting all applicable design basis requirements and retaining the capability to withstand worst case design basis events described in the HC UFSAR, the proposed changes were determined to 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: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

NRC Project Director: William M. Dean.

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

Description of amendment request: The proposed changes revise the descriptive details of Technical Specification 4.7.1.2.1.a, regarding performance testing of the Auxiliary Feedwater (AFW) pumps, to more closely adhere to NUREG–1431, Improved Standard Technical Specifications for Westinghouse Plants. This involves relocating the surveillance-required numerical values for the AFW pump performance test discharge pressure and flow rate to the South Texas Project Updated Final Safety Analysis Report (UFSAR).

Date of amendment request: January 20, 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:

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

The proposed change, which relocates descriptive details (i.e., numerical values for AFW pump discharge pressure and flow rate) of the surveillance testing applicable to the AFW pumps, does not involve a significant increase in the probability or consequences of an accident previously evaluated. The affected AFW pump testing pressure and flow descriptive details that are being removed from SRs 4.7.1.2.1.a.1 and 4.7.1.2.1.a.2 are not related to any assumed initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirement to perform testing on a monthly, staggered basis is not altered by the proposed change, and will remain in the Technical Specifications. The descriptive details of the surveillance testing will be relocated from the Technical Specifications to the USFAR and will be maintained pursuant to 10CFR50.59. The proposed revised wording of SRs 4.7.1.2.1.a.1 and 4.7.1.2.1.a.2 (i.e., to verify the developed head of each pump is greater than or equal to the required developed head) and the relocation of pump testing details to the UFSAR is consistent with the AFW pump test requirements in NUREG–1431. In addition, the surveillance testing details are addressed in existing surveillance procedures that are also controlled by 10CFR50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this proposed 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 relocates descriptive details (i.e., numerical values for AFW pump discharge pressure and flow rate) of surveillance testing applicable to the AFW pumps, which do not meet the criteria for inclusion in Technical Specifications as identified in 10CFR50.36(c)(3). The requirement to perform testing on a monthly, staggered basis is not altered by the proposed change, and will remain in the Technical Specifications. Additionally, relocation of the descriptive testing details is consistent with the wording of the AFW pump test requirements in NUREG–1431, which does not specify minimum numerical pressure and flow limits. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and any future changes to these relocated surveillance testing details or to the applicable surveillance procedures will be evaluated per the requirements of 10CFR50.59. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, 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 relocates descriptive details (i.e., numerical values for AFW pump discharge pressure and flow rate) of the surveillance testing applicable to the AFW pumps, will not reduce a margin of safety since it has no impact on any safety analysis assumptions. The requirement to perform AFW pump testing on a monthly, staggered basis will not be altered by the proposed change, and will remain in the Technical Specifications. Furthermore, the proposed change will not affect the operability requirements of the AFW system as delineated in Specification 3.7.1.2. Since any future changes to these relocated surveillance testing details or to the applicable surveillance procedures will be evaluated per the requirements of 10CFR50.59, there is no reduction in a margin of safety. Finally, this proposed change is also consistent with NUREG–1431, previously approved by the NRC Staff. Revising the Technical Specifications to reflect the approved NUREG–1431 content ensures no significant reduction in the margin of safety. Therefore, this 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon.

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

Date of application for amendments: January 15, 1999 (TS 98–07).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specification (TS) requirements by adding a new action statement to TS 3.1.3.2, ''Position Indicating Systems—Operating,'' that eliminates the need to enter TS 3.0.3 whenever two or more individual rod position indicators (RPIs) may be inoperable per bank, while maintaining the appropriate overall level of protection and adding flexibility to the initial determination of the position of the non-indicating rod(s).

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

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

The proposed change to TS 3.1.3.2 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The potential for the new action statement to impact the probability or consequences of the safety analyses for the plant lies only in the area of operator-exacerbated reactivity events due to a loss of RCCA [rod control cluster assembly] position indication.

RCCA events such as: One or more dropped RCCAs, a dropped RCCA bank or a RCCA ejection (FSAR [Final Safety Analysis Report] Sections 15.2.3 and 15.4.6, respectively) are not impacted since the new action statement does not involve a design change. Events such as: Uncontrolled RCCA bank withdrawal at power, statically misaligned RCCA or withdrawal of a single RCCA (FSAR Sections 15.2.2, 15.2.3, and 15.3.6, respectively) involve, or potentially involve, operator action and are of interest. The uncontrolled RCCA bank withdrawal at power is an ANS [American Nuclear Society] Condition II transient that has been analyzed using a positive reactivity insertion rate greater than that for the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed. Whether the event is caused by a failure in the rod control system or by operator error has no effect on the positive reactivity insertion rate assumed in the analysis. The protection systems assumed in the analysis are unaffected since there is no change to the design. Loss of the RPIS would not result in more frequent control rod movement by plant operators. Therefore, the new action statement would not affect the analysis of this event and departure from nucleate boiling ratio (DNBR) design basis would still be met.

The most severe misalignment situation, with respect to DNBR, arises from cases in which one RCCA is fully inserted or where Bank D is fully inserted to its insertion limits with one RCCA fully withdrawn. For these cases, as discussed in FSAR Section 15.2.3.2, the DNBR remains above the safety analysis limit values. Also, the control bank insertion limit alarms remain available to warn operators that bank insertion limits have been reached.

A compensatory action associated with this new action statement, placing the control rods under manual control, addresses concerns associated with automatic rod motion due to the rod control system and inadvertent operator contribution to these events.

The worst-case event of those described above, the withdrawal of a single RCCA, is an ANS Condition III event. It has been analyzed in FSAR Section 15.3.6, assuming that operators ignore RCCA position indication or that multiple rod control system failures occur. No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from an inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank. This feature is necessary in order to retrieve an accidentally dropped rod. This new action statement does not change the plant design; therefore, there would be no change in the probability of the event being induced by the unlikely, simultaneous electrical failures (FSAR Section 7.7.2.2).

The change in the time to determine the position of the non-indicating rods, indirectly with the movable incore detectors, does not involve a design change nor does it affect the immediate response of the operator to the event, therefore, it does not affect the results of the analyses described above.

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

Since there is no change to the design associated with the proposed change, it does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change involves a loss of the RPIS [Rod Position Indication System] and establishes compensatory measures to maintain control rod position consistent with the assumptions used in the existing accident and transient analyses. The new action statement provides sufficient time for troubleshooting while avoiding unnecessary plant shutdowns per TS 3.0.3.

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

The proposed change to TS 3.1.3.2 does not involve a significant reduction in a margin of safety. As discussed in Section IV.A above, the results of the FSAR Chapter 15 safety analyses for the applicable events, are not affected by the proposed changes. Therefore, the safety margins demonstrated by these analyses remain unchanged. The additional time to obtain the flux maps is consistent with the 12-hour time frame allowed to verify shutdown margin when a rod is misaligned from its group step counter height by more than plus or minus 12 steps in TS 3.1.3.1 and remains within a shiftly basis. Therefore, it does not reduce the margin of safety.

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

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

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

NRC Project Director: Cecil O. Thomas.

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 amendment request: January 29, 1999 (TSCR 211).

Description of amendment request: The proposed amendments reflect changes to sections 15.6 and 15.7 of the Point Beach Nuclear Plant (PBNP), Units 1 and 2, Technical Specifications (TS). The proposed 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.

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 Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of an accident previously evaluated.

These changes are administrative and therefore do not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions. The proposed TS changes do not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the administrative changes described above.

In addition, initiating conditions and assumptions are unchanged and remain as previously analyzed for accidents in the PBNP Final Safety Analysis Report. The proposed TS changes do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated. All Limiting Conditions [for] Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, these changes do not increase the probability of previously evaluated accidents.

These changes do not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy and operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, these changes do not affect the consequences of previously evaluated accidents.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

These changes do not introduce nor increase the number of failure mechanisms of a new or different type than those previously evaluated since there are no physical changes being made to the facility. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not introduced.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant reduction in the margin of safety because existing component redundancy is not being changed by these proposed changes. There are no new or significant changes to the initial conditions contributing to accident severity or consequences, and safety margins established through the design and facility license including the Technical Specifications remain unchanged. Therefore, there are no significant reductions in a margin of safety introduced by [these] proposed amendment[s].

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: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: December 29, 1998.

Description of amendment request: This amendment would revise the Wolf Creek Technical Specification (TS) Figures 3.4–2, 3.4–3, and 3.4–4 to incorporate revised reactor coolant system heatup and cooldown limit curves and a revised cold overpressure mitigation system (COMS) power operated relief valve (PORV) setpoint limit curve.

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.

Incorporating the revised heatup and cooldown pressure/temperature limit curves and the COMS PORV setpoint limit curve into the WCGS Technical Specifications does not affect the probability or consequences of an accident previously evaluated.

The revised limit curves are calculated using the most limiting RT_{NDT} for the reactor vessel components and include a radiation-induced shift corresponding to the end of the period for which the curves are generated. The COMS PORV Setpoint Limit Curve is calculated using the most limiting mass injection transient, taking into account operation of the NCP [normal charging pump] during shutdown modes. The changes do not affect the basis, initiating events, chronology, or availability/operability of safety related equipment required to mitigate transients and accidents analyzed for WCGS.

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

Adopting the revised limit curves redefines the range of acceptable operation for the Reactor Coolant System. This redefinition is a result of the analysis of reactor vessel surveillance specimens removed from the reactor in a continuing surveillance program which monitors the effects of neutron irradiation on the WCGS reactor vessel materials under actual operating conditions. Included in the revised limit curves is consideration for NCP operation during shutdown modes. Incorporating these revised curves does not create the possibility of an accident of a different type from any previously evaluated for WCGS.

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

The revision of these limit curves continues to maintain the margin of safety required for prevention of non-ductile failure of the WCGS reactor vessel during low temperature operation as required by 10 CFR 50, Appendices G and H. The revised curves primarily affect RCS [reactor coolant system] operation below 350°F by limiting the available pressure/temperature window for heatup and cooldown. The revised limit curves compensate for the in-service radiation induced embrittlement of the reactor vessel and accounts for the requirement that the closure flange region temperature must exceed the nil-ductility temperature by at least 120°F when pressure exceeds 20% of the preservice hydrostatic test pressure.

The revised COMS PORV Setpoint Limit Curve, which includes consideration of NCP operation during shutdown modes, ensures overpressure protection of the RCS and reactor vessel.

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: January 12, 1999.

Description of amendment request: This license amendment request proposes to revise Wolf Creek Generating Station (WCGS) Technical Specification 3/4.7.5, Ultimate Heat Sink, to add a new action statement. Specifically, the new action statement will require verification of operability of the two residual heat removal (RHR) trains, or initiation of power reduction with only one RHR train operable, when the plant inlet water temperature is between 90 and 94 degrees Farenheit. The current TS requires shutdown when plant inlet water temperature exceeds 90 degrees Farenheit.

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 does not involve any physical alteration of plant systems, structures or components. The proposed change provides an allowed time for the plant to continue operation with plant inlet water temperature in excess of the current technical specification limit of 90 degrees Fahrenheit, up to 94 degrees Fahrenheit, which is less than the design limit of 95 degrees Fahrenheit for plant components. The plant inlet water temperature is not assumed to be an initiating condition of any accident analysis evaluated in the updated safety analysis report (USAR). Therefore, the allowance of a limited time for the water temperature to be in excess of the current limit does not involve an increase in the probability of an accident previously evaluated in the USAR. The UHS [ultimate heat sink] supports operability of safety related systems used to mitigate the consequences of an accident. Plant operation for brief periods with plant inlet water temperature greater than 90 degrees Fahrenheit up to 94 degrees Fahrenheit will not adversely affect the operability of these safety-related systems and will not adversely impact the ability of these systems to perform their safety-related functions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the USAR.

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

The proposed change does not involve any physical alteration of plant systems, structures or components. The temperature of the plant inlet water being greater than 90 degrees Fahrenheit but less than or equal to 94 degrees Fahrenheit for a short period does not introduce new failure mechanisms for systems, structures or components not already considered in the USAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change will allow an increase in plant inlet water temperature above the current technical specification limit of 90 degrees Fahrenheit for the Ultimate Heat Sink, and delay the requirement to shutdown the plant when the plant inlet water system temperature limit is exceeded for 12 hours. The proposed change does not alter any safety limits, limiting safety system settings, or limiting conditions for operation, and the proposed temperature increase will remain below the design limit cooling water input value for safety-related equipment, except for the unlikely event of a combination of a worst dam failure occurring with a loss of coolant accident during a period of severe meteorological conditions. Thus, the proposed change does not involve a significant reduction in any margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

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

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

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: January 22, 1999.

Brief description of amendment: The amendment would revise Technical Specification Surveillance Requirement 3.8.1.7 to better match plant conditions during diesel generator (DG) testing by clarifying which voltage and frequency limits are applicable during the transient and steady state portions of the DG start.

Date of publication of individual notice in **Federal Register**: February 1, 1999 (64 FR 4902).

Expiration date of individual notice: March 3, 1999.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401.

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

Date of application for amendment: January 20, 1999.

Brief description of amendment request: The proposed amendment requests changes to the Technical Specification degraded voltage relay setpoints.

Date of publication of individual notice in **Federal Register**: January 28, 1999 (64 FR 4474).

Expiration date of individual notice: March 1, 1999.

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

PP&L, Inc., Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: November 23, 1998.

Brief description of amendment request: The requested changes would change the allowable values for both the core spray system and the low pressure coolant injection system reactor steam dome pressure-low functions.

Date of publication of individual notice in **Federal Register**: February 1, 1999 (64 FR 4904).

Expiration date of individual notice: March 3, 1999.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: November 30, 1998, as supplemented by letter dated January 8, 1999.

Brief description of amendments: The amendments relocate the requirement for removal of the Reactor Protection System (RPS) shorting links to the Updated Final Safety Analysis Report (UFSAR).

Date of issuance: February 8, 1999.

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

Amendment Nos.: 170; 165 & 183; 180.

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

Date of initial notice in **Federal Register**: January 7, 1999. (64 FR 1032).

The January 8, 1999, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 27, 1998 (NRC–98–0033).

Brief description of amendment: The amendment revises technical specifications (TS) 3.5.2 and 3.5.3 and the associated Bases, raising the minimum water level for the core spray system in the condensate storage tank (CST). The amendment also removes incorrect information from TS 3.5.3 regarding water inventory in the CST reserved for the high pressure coolant injection and reactor core isolation cooling systems.

Date of issuance: February 8, 1999.

Effective date: February 8, 1999, with full implementation within 90 days.

Amendment No.: 131.

Facility Operating License No. NPF–43. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19967).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of amendment request: March 25, 1998, as supplemented by letter dated November 30, 1998.

Brief description of amendment: The amendment changes the Appendix A Technical Specifications (TSs) by modifying TS 3.9.8.1, ''Shutdown Cooling and Coolant Circulation-High water Level,'' and TS 3.9.8.2, ''Shutdown Coolant Circulation-Low Water Level,'' to change the minimum water level above the fuel assemblies seated in the reactor vessel at which the Shutdown Cooling System (SDC) is required to be maintained operable, or be in operation. Also TS 3.8.1.2, ''Electric Power Systems A.C. Sources Shutdown,'' and appropriate Bases are revised to make wording consistent with the TS 3.9.8.1 and 3.9.8.2.

Date of issuance: February 2, 1999.

Effective date: This license amendment is effective as of its date of issuance, to be implemented within 60 days.

Amendment No.: 148.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25109).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 27, 1998.

Brief description of amendment: This amendment revises TS 3/4.8.2.3, ''Electrical Power Systems—DC Distribution—Operating,'' and the associated bases. The surveillance requirements for battery testing have been revised.

Date of issuance: February 9, 1999.

Effective date: February 9, 1999.

Amendment No.: 229.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64125).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 30, 1998, as supplemented on December 9, 1998.

Brief description of amendment: This amendment revised Technical Specification 3.1.7, ''Standby Liquid Control System,'' by increasing the boron concentration in the Standby Liquid Control System for Cycle 8 fuel design.

Date of issuance: February 8, 1999.

Effective date: February 8, 1999.

Amendment No.: 97.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40562).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 27, 1998.

Brief description of amendments: The amendments revised Turkey Point Units 3 and 4 Technical Specifications to add the qualifications for the multi-discipline supervisor.

Date of issuance: February 3, 1999.

Effective date: February 3, 1999.

Amendment Nos.: 199 and 193.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR 69341).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Florida International

University, University Park, Miami, Florida 33199.

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

Date of application for amendment: July 21, 1998, as supplemented October 6, December 16, and December 31, 1998.

Brief description of amendment: The amendment changes various Reactor Protection System (RPS) and Engineered Safety Feature Actuation System setpoints and allowable values; corrects the specified maximum reactor power level limited by the high power level RPS trip; adds a new Technical Specification associated with the automatic isolation of steam generator blowdown; and makes several editorial changes to correct various errors and to provide needed clarification. The amendment also makes changes to the applicable Bases pages and expands the Bases to discuss the new requirements for the automatic isolation of steam generator blowdown. However, the staff has not completed its evaluation of the requested change in the trip setpoint and allowable values for the steam generator water level. This portion of the request will be addressed later.

Date of issuance: February 8, 1999.

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

Amendment No.: 226.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43208).

The October 6, December 16, and December 31, 1998, letters provided clarifying information that did not change the scope of the July 21, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 3, 1998, as supplemented May 7, 1998.

Brief description of amendment: The amendment revises the Millstone Unit 3 licensing basis by eliminating the requirement to have the recirculation spray system directly inject into the reactor coolant system following a design-basis accident, with the exception of loss-of-coolant accident (LOCA) scenarios involving a long-term passive failure. The Millstone Unit 3 licensing basis maintains the direct injection requirement for scenarios, as a contingency, for situations where it may be needed—as in the case of a LOCA with a long-term passive failure or for beyond design-basis scenarios.

Date of issuance: January 20, 1999.

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

Amendment No.: 165.

Facility Operating License No. NPF–49: Amendment revised the Millstone Unit 3 licensing basis.

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14487).

The May 7, 1998, letter provided clarifying information that did not change the scope of the March 3, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment and final no significant hazards consideration determination are contained in a Safety Evaluation dated January 20, 1999.

No significant hazards consideration comments received: No public comments received.

A petition to intervene was received from the Citizens Regulatory Commission that was dismissed and terminated by the NRC Atomic Safety Licensing Board.

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

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: April 16, 1998.

Brief description of amendment: The amendment changes the Technical Specifications to modify a testing requirement for the emergency diesel generators.

Date of issuance: February 9, 1999.

Effective date: February 9, 1999.

Amendment No.: 187.

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

Date of initial notice in **Federal Register**: October 21, 1998, (63 FR 56256).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 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 10601.

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: June 16, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 6 to relocate the Safety Review Committee Reviews, Audits and Records from TS to the Quality Assurance Program Section of the Final Safety Analysis Report.

Date of issuance: February 8, 1999.

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

Amendment No.: 251.

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

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38204).

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

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: October 19, 1998.

Brief description of amendment: This amendment eliminates restrictions imposed by Technical Specification (TS) 3.0.4 for the Filtration, Recirculation and Ventilation System during fuel movement and CORE ALTERATION activities. Specifically, TS Limiting Conditions for Operation 3.6.5.3.1 and 3.6.5.3.2 have been revised to add a note stating that the provisions of TS 3.0.4 are not applicable for initiation of handling of irradiated fuel in the secondary containment and CORE ALTERATIONS provided that the plant is in OPERATIONAL CONDITION 5, with reactor water level equal to or greater than 22 feet 2 inches.

Date of issuance: February 4, 1999.

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

Amendment No.: 113.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 4121).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: September 8, 1998, as supplemented December 8, 1998.

Brief description of amendment: This amendment revised Appendix C, ''Additional Conditions,'' and will allow the performance of single cell charging and the use of non-Class 1E single cell battery chargers, with proper electrical isolation, for charging connected cells in OPERABLE Class 1E batteries. The single cell chargers will be used to restore individual cell parameters to the normal limits specified in Technical Specification Table 4.8.2.1–1.

Date of issuance: February 9, 1999.

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

Amendment No.: 114.

Facility Operating License No. NPF–57: This amendment revised Appendix C of the license.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53954).

The December 8, 1998, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: April 28, 1998, as supplemented September 29, 1998, and December 8, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.4.2.1 to replace the ±1% setpoint tolerance limit for safety/relief valves (SRVs) with a ±3% setpoint tolerance limit. In addition, the amendment revises TS 4.4.2.2 to state that all SRVs will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing.

Date of issuance: February 10, 1999.

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

Amendment No.: 115.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33108).

The September 29, 1998, and December 8, 1998, supplements provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: June 25, 1998, as supplemented August 25, 1998, and December 15, 1998.

Brief description of amendment: The amendment revised Technical Specification (TS) Surveillance Requirement 4.5.1.d.2.b by deleting the requirement to perform in-situ functional testing of the Automatic Depressurization System safety relief valves (SRVs) during startup testing activities. The amendment also revised TS Surveillance Requirement 4.4.2.1 such that the 18-month channel calibration for the SRV acoustic monitors will no longer require an exception to the provisions of TS 4.0.4, nor adjustments to SRV full open noise levels.

Date of issuance: February 10, 1999.

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

Amendment No.: 116.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43212).

The August 25, 1998, and December 15, 1998, supplements provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

Date of application for amendment: October 12, 1998.

Brief description of amendment: This amendment allowed a one-time extension of the Technical Specification (TS) surveillance interval to the end of fuel Cycle 10 for certain TS surveillance requirements (SRs). Specifically, the amendment extended the surveillance interval in (a) SR 4.3.2.1.3 for the instrumentation response time testing of each engineered safety features actuation system function, (b) SRs 4.8.2.3.2.f and 4.8.2.5.2.d for service testing of the 125-volt DC and the 28-volt DC distribution system batteries, respectively, and (c) SR 4.8.2.5.2.c.2 for verification that the 125-volt DC battery connections are clean, tight, and coated with anti-corrosion material. Because of the length of the last outage and delays in restart, the SRs would have become overdue prior to reaching the next refueling outage (2R10). The SRs are to be completed during the 2R10 outage, prior to returning the unit to Mode 4 (hot shutdown) upon outage completion.

Date of issuance: February 1, 1999.

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

Amendment No.: 198.

Facility Operating License No. DPR–75: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 4, 1998 (63 FR 59594).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 1, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 14, 1997, as supplemented by letters dated March 13, 1998, and November 10, 1998.

Brief description of amendments: The amendments would revise the licensing basis as described in Section 3.5, ''Missile Protection,'' of the Updated Final Safety Analysis Report to allow the use of NUREG–0800, ''Standard Review Plan'' methodology in evaluating tornado-generated missiles.

Date of issuance: February 9, 1999.

Effective date: February 9, 1999, to be implemented in the next periodic update of the Updated Final Safety Analysis Report (UFSAR) in accordance with 10 CFR 50.71(e) that occurs after 60 days of the date of issuance.

Amendment Nos.: Unit 2—148; Unit 3—140.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the UFSAR.

Date of initial notice in **Federal Register**: December 31, 1997 (62 FR 68315).

The March 13, 1998, and November 10, 1998, supplemental letters provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 26, 1998, as supplemented by letters dated September 18 and November 30, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) as follows: (1) The Applicability of Limiting Condition for Operation (LCO) 3.3.6, ''Containment Ventilation Isolation Instrumentation,'' is revised to refer to TS Table 3.3.6–1; the TS table is revised to add a column entitled ''Applicable Modes or Other Specified Conditions.'' Then, the applicable modes for Manual Initiation, Automatic Actuation Logic and Actuation Relays, and Safety Injection are revised to include only Modes 1, 2, 3, and 4. Consistent with this change, LCO 3.3.6, Condition C and Required Action C.2 are revised to reflect that system level manual initiation and automatic actuation are not required during core alterations and/or during movement of irradiated fuel assemblies within containment. Appropriate Bases changes are included to reflect the TS changes. (2) LCO 3.9.4 is revised to allow the emergency air lock to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. In addition, the LCO statement is revised to reflect that containment ventilation isolation (CVI) would be accomplished by manually closing the individual containment purge supply and exhaust isolation valves as opposed to a system level manual or automatic initiation, consistent with the proposed change to LCO 3.3.6. Surveillance Requirement (SR) 3.9.4.2 is revised to reflect the change to CVI. Appropriate Bases changes are included to reflect the TS changes. (3) LCO 3.7.6 is revised to delete the words ''Redundant CSTs'' from the title and LCO 3.7.6a is deleted. Appropriate Bases changes are included to reflect the changes.

Date of issuance: January 29, 1999.

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

Amendment Nos.: Unit 1—105; Unit 2—83.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53955). The supplement dated November 30, 1998, provided clarifying information that did not change the scope of the application and the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 13, 1998, as supplemented by letters dated December 16, 1998, and January 13, 1999.

Brief description of amendments: The amendments revise Technical Specification Section 1.1, Definitions, for ''Engineered Safety Feature [ESF] Response Time'' and ''Reactor Trip System [RTS] Response Time'' to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

Date of issuance: February 8, 1999.

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

Amendment Nos.: 106 and 84.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53957).

The December 16, 1998, and January 13, 1999, letters provided clarifying information that did not change the scope of the July 13, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 1999.

No significant hazards consideration comments received: No.

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

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: October 29, 1998.

Brief description of amendments: Relocates portions of Technical Specification 4.8.1.1.2.g requirements regarding maintenance of the diesel generator fuel oil storage tank to the Technical Requirements Manual.

Date of issuance: February 8, 1999.

Effective date: The license amendment is effective as of its date of issuance, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 102; Unit 2—Amendment No. 89.

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

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR 69347).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 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, TX 77488.

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

Date of application for amendments: November 16, 1998.

Brief description of amendments: The amendments revise the Sequoyah Nuclear Plant Technical Specification (TS) emergency diesel generator surveillance requirements. The U.S. Nuclear Regulatory Commission staff has found the proposed changes to be acceptable.

Date of issuance: February 9, 1999.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 242 and 232.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TSs.

Date of initial notice in **Federal Register**: December 2, 1998 (63 FR 66603).

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

No significant hazards consideration comments received: No.

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

Yankee Atomic Electric Company, Docket No. 50–29, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of application for amendment: August 20, 1998.

Brief description of amendment: Revises Technical Specifications (TS) through deletion of definition of SITE BOUNDARY, moves site map from TS to Final Safety Analysis Report and deletion of an uneeded reference to the site map.

Date of issuance: February 3, 1999.

Effective date: February 3, 1999.

Amendment No.: 150.

Possession Only License No. DPR–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53962). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Dated at Rockville, Maryland, this 17th day of February 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–4391 Filed 2–23–99; 8:45 am]

BILLING CODE 7590–01–P

PRESIDIO TRUST

Notice of Public Meeting

AGENCY: The Presidio Trust.

ACTION: Notice of public meeting.

SUMMARY: In accordance with § 103(c)(6) of the Presidio Trust Act, 16 U.S.C. 460bb note, Title I of Pub. L. 104–333, 110 Stat. 4097, and in accordance with the Presidio Trust's bylaws, notice is hereby given that a public meeting of the Presidio Trust Board of Directors will be held from 10:00 a.m. to 12:00 p.m. (PST) on Wednesday, March 24, 1999, at the Presidio Golden Gate Club, Fisher Loop, Presidio of San Francisco, California. The Presidio Trust was created by Congress in 1996 to manage approximately eighty percent of the former U.S. Army base known as the Presidio, in San Francisco, California.

The purposes of this meeting are: (i) to consider presentations from the four finalists for the ground lease and development of the Letterman Complex and, possibly, (ii) to present an update regarding restoration activities at Crissy Field. Public comment on these topics will be received and memorialized in accordance with the Trust's Public Outreach Policy.

TIME: The meeting will be held from 10:00 a.m. to 12:00 p.m. (PST) on Wednesday, March 24, 1999.

ADDRESSES: The meeting will be held at the Presidio Golden Gate Club, Fisher Loop, Presidio of San Francisco.

FOR FURTHER INFORMATION CONTACT: Karen A. Cook, General Counsel, the Presidio Trust, 34 Graham Street, P.O. Box 29052, San Francisco, California 94129–0052, Telephone: (415) 561–5300.

Dated: February 18, 1999.

Karen A. Cook,

General Counsel.

[FR Doc. 99–4510 Filed 2–23–99; 8:45 am]

BILLING CODE 4310–4R–P

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting Notice

FEDERAL REGISTER CITATION OF PREVIOUS ANNOUNCEMENT: [64 FR 7930–7931, February 17, 1999].

STATUS: Closed Meeting.

PLACE: 450 Fifth Street, N.W., Washington, D.C.

DATE PREVIOUSLY ANNOUNCED: February 17, 1999.

CHANGE IN THE MEETING: Deletion.

The following item was not considered at the closed meeting held on Thursday, February 18, 1999, at 11:00 a.m.:

Formal order of investigation.

Commissioner Carey, as duty officer, determined that Commission business required the above change and that no earlier notice thereof was possible.

At times, changes in Commission priorities require alterations in the scheduling of meeting items. For further information and to ascertain what, if any, matters have been added, deleted or postponed, please contact:

The Office of the Secretary (202) 942–7070.

Dated: February 19, 1999.

Jonathan G. Katz,

Secretary.

[FR Doc. 99–4700 Filed 2–22–99; 12:12 pm]

BILLING CODE 8010–01–M

SECURITIES AND EXCHANGE COMMISSION

Golden Mountain, Inc.; Order of Suspension of Trading

[File No. 500–1]

February 22, 1999.

It appears to the Securities and Exchange Commission that there is a lack of current, adequate and accurate information concerning the securities of Golden Mountain, Inc., a Nevada shell