1998, NRC published the schedule for these workshops and indicated that a workshop would be held on October 20–21, 1999, at NRC Headquarters at Two White Flint North, 11545 Rockville Pike, Rockville, MD. At the conclusion of the August workshop, the participants agreed to postpone the October workshop until February 2000. The rescheduling will allow more time for public review of the SRP prior to the final workshop. The workshop will be held at NRC Headquarters at Two White Flint North, 11545 Rockville Pike, Rockville, MD. NRC staff will announce the date for this workshop in a future **Federal Register**.

FOR FURTHER INFORMATION CONTACT: Dominick A. Orlando, Decommissioning Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards, at (301) 415–6749.

Dated at Rockville, Maryland, this 30th day of August 1999.

For the US Nuclear Regulatory Commission.

Larry W. Camper,

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

[FR Doc. 99–23298 Filed 9–7–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of September 6, 13, 20, 27 *and* October 18, 1999.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of September 6

Tuesday, September 7

9:15 a.m.—Briefing on PRA Implementation Plan (Public Meeting) (Contact: Tom King, 301–415–5790).

Friday, September 10

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

a. Final Rule: ''Respiratory Protection and Controls to Restrict Internal Exposures, 10 CFR Part 20''

b. Yankee Atomic Electric Company (Yankee Nuclear Power Station), Docket No. 50–029–LA, Yankee Atomic's Motion for Leave to Withdraw Appeal of LBP–99–14

Week of September

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

Week of September 20—Tentative

Tuesday, September 21

9:25 a.m.—Affirmation Session (Public Meeting), (if needed).

9:30 a.m.—Briefing by DOE on Draft Environmental Impact Statement (DEIS) for a Proposed HLW Geologic Repository (Public Meeting).

Wednesday, September 22

9:00 a.m.—Meeting on Center for Strategic and International Studies Report, ''The Regulatory Process for Nuclear Power Reactors—a Review'' (Public Meeting).

Week of September 27—Tentative

There are no meetings scheduled for the Week of September 27.

And

Week of October 18—Tentative

Thursday, October 21

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

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

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

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

Dated: September 3, 1999.

William M. Hill, Jr.,

Secy, Tracking Officer, Office of the Secretary.

[FR Doc. 99–23425 Filed 9–3–99; 2:36 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 14, 1999, through August 27, 1999. The last biweekly notice was published on August 25, 1999 (64 FR 46424).

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 October 8, 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.

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

Date of amendment request: August 3, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 2.1.B to increase the minimum critical power ratio for higher cycle exposures for Unit 2. The proposed amendments would also revise TS 6.9.A.6.b for Units 2 and 3 to add an NRC-approved topical report to the list of analytical methodologies that are used to determine operating limits.

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

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

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

Changing the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) at Dresden Nuclear Power Station Unit 2 will not increase the probability or the consequences of an accident previously evaluated. This change implements the MCPR SL resulting from the Siemens Power Corporation (SPC) ANFB critical power correlation methodology using the approved ATRIUM–9B additive constant uncertainty. For each cycle, specific MCPR SL calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR SL. Additionally, operational MCPR limits will be applied that will ensure the MCPR SL is not violated during all modes of operation and anticipated operational occurrences. The MCPR SL ensures that less than 0.1% of the rods in the core are expected to experience boiling transition. Therefore, the probability or consequences of an accident will not increase.

Adding EMF–85–74, Revision 0, Supplements 1 and 2 (P)(A) to Section 6 for Dresden Nuclear Power Station Units 2 and 3, does not increase the probability or consequences of an accident previously evaluated. The NRC-approved burnup extension for RODEX2A applications has been demonstrated to meet all applicable design criteria. Therefore, adding this methodology to Technical Specification Section 6 does not increase to the probability or consequences of an accident previously evaluated.

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

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

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

The MCPR SL is changing for Dresden Nuclear Power Station Unit 2 to support Cycle 17 operation. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

The addition of RODEX2A (EMF–85–74, Revision 0, Supplements 1 and 2 (P)(A)) to Section 6 does not create the possibility of a new accident from an accident previously evaluated. This change does not alter or add any new equipment or change modes of operation. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

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

Changing the MCPR SL for Dresden Nuclear Power Station Unit 2 will not involve any reduction in margin of safety. The MCPR SL provides a margin of safety by ensuring that less than 0.1% of the rods are calculated to be in boiling transition. The proposed Technical Specification amendment request reflects the MCPR SL results from evaluations by SPC using NRC-approved methodology.

Because the methodology used to determine the MCPR SL is conservative and has received NRC approval, a decrease in the margin to safety will not occur due to changing the MCPR SL. The revised MCPR SL will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR SL to ensure that the MCPR SL is not violated during all applicable modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

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

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: August 13, 1999, as supplemented on August 27, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification Section 1.0, ''Definitions,'' Item 1.7, ''Core Alteration,'' to specify that movement of instrumentation and control rod movements are not considered core alterations if there are no fuel assemblies in the associated cell. The licensee also proposed corresponding changes to TS Sections 3/4.1, 3/4.3, and 3/4.9 to reflect the change in definition.

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

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

The proposed changes incorporate a definition contained in NUREG–1433, Revision 1, ''Standard Technical Specifications, General Electric Plants, BWR/4.'' There are no modifications to plant equipment or systems and there is no direct effect on plant operation. The proposed changes do not affect any accident initiators or precursors and do not change or alter the design assumptions for systems or components used to mitigate the consequences of an accident. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. The proposed changes do not impact the requirements for refueling evolutions associated with shutdown margin, core monitoring, and reactor protection system operability. There are no changes to parameters governing plant operation, and no new or different types of equipment will be installed. These changes do not impact any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). Therefore, no increases in the probability of an accident or consequences will result due to this change.

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

The proposed changes do not affect the design or operation of any plant system, structure, or component. There are no changes to parameters governing plant operation, and no new or different type of equipment will be installed. There is no change in any method by which a safety related system performs its function. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints affected by this proposed action. This proposed action will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. As such, no new failure modes are being introduced. There are no changes to assumptions in accident analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are consistent with NUREG–1433, Revision 1, ‘‘Standard Technical Specifications, General Electric Plants, BWR/4.’’ The proposed changes do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The initial conditions and methodologies used in the accident analyses remain unchanged. Therefore, accident analyses results are not impacted. There are no resulting effects on plant safety parameters or setpoints. The proposal does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, these proposed changes do not cause a reduction in the margin of safety.

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

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

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

NRC Section Chief: Anthony J. Mendiola.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: May 5, 1999.

Description of amendment request: The proposed amendment would permit a one-time extension of the allowed outage time (AOT) for the reactor protection and engineered safety feature actuation instrumentation.

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

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

The reactor protection and engineered safety features functions are not initiators of any design basis accident or event and therefore do not increase the probability of any accident previously evaluated. The proposed changes to the AOTs, bypass times, and allowing on-line testing and maintenance have an insignificant impact on plant safety based on the calculated CDF [core damage frequency] increase being less than LOE–06. Therefore, the proposed changes do not result in a significant increase in the 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 do not result in a change in the manner in which the RPS [reactor protection system] and ESFAS [engineered safety features actuation system] provide plant protection. No change is being made which alters the functioning of the RPS and ESFAS. Rather, the likelihood or probability of the RPS or ESF functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident nor involve a reduction in the margin of safety as defined in the Safety Analysis Report.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operations are determined. The impact of increased AOTs, testing times, and allowing on-line testing and maintenance are expected to result in an overall improvement in safety because:

The longer AOTs for the master relays, logic cabinets, and analog channels will promote improved maintenance practices that will provide improved component performance, improved availability of the protection system, and a reduced number of spurious reactor trips and spurious actuation of safety equipment.

The longer AOTs and bypass times for the analog channels will provide additional time before being required to place the channel in trip. With the channel in trip, the logic required to cause a reactor trip or a safety system actuation is reduced to 1 of 2 (for 2 of 3 logic) and to 1 of 3 (for 2 of 4 logic). With the reduced logic requirement, the potential for a spurious actuation is increased. Leaving the channel in the bypass state for additional time does reduce the availability of signals to initiate component actuation for event mitigation when required, but as shown in this analysis, the impact on plant safety is small due to the availability of other signals or operator action to trip the reactor or cause component actuation.

The longer allowed outage times will provide plant operators additional flexibility in operating the plant. There will be additional time available before an action needs to be taken to shut down the plant or place a channel in the tripped state. This additional flexibility will facilitate prioritizing component repairs.

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 10610. Biweekly Notice Coordinator Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: August 4, 1999.

Description of amendment request: The amendments would revise the joint Technical Specifications as follows:

(1) A current action in Section 3.2.2 requires that when one Nuclear Service Water System (NSWS) suction transfer low pit level channel is inoperable, the channel be placed in its trip position. The licensee proposed an additional alternative such that the NSWS suction can simply be aligned from Lake Wylie to the Standby Nuclear Service Water Pond (SNSWP). Suction from Lake Wylie is the normal configuration, while suction from the SNSWP is the safety configuration. This proposed alternative

action provides operational flexibility; there is no associated design change to the units.

(2) The licensee proposed to delete from Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' the entry regarding Auxiliary Feedwater Loss of Offsite Power (Function 6d) on the basis that a comparable and adequate requirement will exist in Section 3.3.5. To such end, a new Surveillance Requirement (SR) 3.3.5.3 will be added, incorporating the Function 6d requirement from Table 3.3.2–1. These proposed changes remove inconsistencies that currently exist in the Technical Specifications for Function 6d. There is no associated design change to the units.

(3) In the process of converting the Technical Specification to the improved format (Amendment Nos.173 and 165), errors were inadvertently introduced regarding the conditions under which the Reactor Coolant System Subcooling Margin Monitor must be operable. The licensee proposed to correct these errors by revising the entry regarding the Subcooling Margin Monitor in Table 3.3.3–1, ''Post Accident Monitoring Instrumentation''. There is no associated design change to the units.

(4) Section 3.4.17 is concerned with reactor coolant system loops test exceptions. Currently Surveillance Requirement 3.4.17.2 incorrectly specifies that a COT [channel operational test] be performed ''for each power range neutron flux-flow and intermediate range neutron flux channel and P–7 [Low Power Reactor Trips Block Function]''. The licensee proposed to correct this statement by deleting ''P–7'' and adding ''P–10 [Power Range Neutron Flux] and P–13 [Turbine Impulse Pressure]''. This correction does not involve any design change to the units.

(5) The licensee proposed to delete from Section 5.3.1 the specific qualification requirements for Reactor Operators (ROs) and Senior Reactor Operators (SROs). Such requirements are specified by 10 CFR 50.55, ''Operators'' Licenses'', and the licensee is required to follow this regulation. There will be no change in the qualification of ROs and SROs, and no design change to the units.

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

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. For proposed changes #1–4, the systems and equipment referenced in the revised TS are not accident initiating systems; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the systems is not being modified by these proposed changes. Therefore, there will be no impact on any accident consequences. For proposed change #5, the change is purely administrative; it will therefore have no effect on any accident probabilities or 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 adversely 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 systems and equipment referenced in the revised TS for proposed changes #1–4 are already capable of performing as designed. No safety margins will be impacted. Since proposed change #5 is purely administrative, it will have no effect on any safety margins.

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

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: July 26, 1999.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' to remove the ''Trip Setpoint'' values and revise the ''Allowable Values'' entries for Sequence Logic Channels a, ''Essential Bus Feeder Breaker Trip (90%),'' and b, ''Diesel Generator Start, Load Shed on Essential Bus (59%).''

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes do not change any accident initiator, initiating condition, or assumption.

The proposed changes would revise Technical Specification (TS) Table 3.3–4, Safety Features Actuation System Instrumentation Trip Setpoints, to remove the'Trip Setpoint'' values for Functional Unit Sequence Logic Channel ''a'', ''Essential Bus Feeder Breaker Trip (90%)'', and Functional Unit Sequence Logic Channel ''b'', ''Diesel Generator Start, Load Shed on Essential Bus (59%)'', and also modify the ''Allowable Values'' entry for Functional Unit Sequence Logic Channel ''a'', consistent with updated calculations and current setpoint methodology. The proposed changes would also clarify an inconsistency between Table 3.3–4 and Table 4.3–2, Safety Features Actuation System Instrumentation Surveillance Requirements. The proposed changes to Limiting Condition for Operation (LCO) 3.3.2.1 and Bases 3/4.3.1 and 3/4.3.2 are associated with these changes.

The accident previously evaluated in Section 15.2.9, ''Loss of All AC Power to the Station Auxiliaries (Station Blackout),'' of the DBNPS Updated Safety Analysis Report (USAR) is not affected by the proposed changes because its bounding conditions are not affected. The existing TS action statements will continue to maintain the USAR requirement to start and load one Emergency Diesel Generator (EDG) to meet minimum ESF requirements, should all AC power be lost. Furthermore, the proposed changes are based on the existing performance characteristics of plant equipment; therefore, the proposed changes will not involve a significant change to the plant design or operation.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path or alter radiological consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety because the proposed changes do not significantly reduce the ability of the plant to respond to a loss of AC power to the essential 4160 Volt buses in a timely manner. The revised Allowable Value for the Sequence Logic Channel ''Essential Bus Feeder Breaker Trip (90%)'' takes into account the need not only to be able to actuate Engineered Safety Features equipment coincident with a degraded grid condition, but to provide voltage at the required value to properly operate the equipment.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: July 27, 1999.

Description of amendment request: The proposed amendment would remove Technical Specification (TS) Section 6.4, ''Training,'' relocate TS Sections 6.5.2.8, ''Audits,'' and 6.10 ''Record Retention,'' to the Updated Safety Analysis Report, and make related changes to TS Sections 6.14, ''Process Control Program,'' and 6.15, ''Offsite Dose Calculation Manual.'' In addition, an editorial correction is proposed to TS 6.8, ''Procedures and Programs.''

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are affected by the proposed changes to Section 6.0, Administrative Controls, of the Technical Specifications (TS).

The proposed changes to remove Section 6.4, Training, from the TS and relocate the detailed listings of TS Section 6.5.2.8, Audits, and TS Section 6.10, Record Retention, to the DBNPS [Davis-Besse Nuclear Power Station] Quality Assurance Program in Chapter 17 of the Updated Safety Analysis Report are consistent with NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 1 or NRC Administrative Letter 95–06 ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance,'' dated December 12, 1995. The proposed changes to TS Section 6.14, Process Control Program (PCP); TS Section 6.15, Offsite Dose Calculation Manual (ODCM); and TS Section 6.8, Procedures and Programs, are either associated administratively with the above proposed changes or are editorial corrections. These TS being removed or relocated will remain subject to the controls of regulations (e.g., 10 CFR 50.59, 10 CFR 55.59, or 10 CFR 50.54(a)).

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. As described above, these changes are consistent with the improved ''Standard Technical Specifications—Babcock and Wilcox Plants'' (NUREG–1430) or Administrative Letter 95–06 and are administrative changes. The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes, which involve only administrative controls. The proposed changes do not alter any accident scenarios.

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative and do not reduce or adversely affect the capabilities of any plant structures, systems or components to perform their nuclear safety function.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: August 5, 1999.

Description of amendment request: The requested changes correct editorial errors in Technical Specification (TS) Sections 3.8.3.2, 4.6.2.1, 4.6.2.2, 4.8.1.1, and 4.9.12. Also, the requested changes correct minor editorial and reference errors in Technical Specification Bases Sections B 3/4.3.2, B 3/4.4.11, B 3/4.6.1.2, and B 3/4.8.4.

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:

NNECO [Northeast Nuclear Energy Company] has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve any Significant Hazards Considerations (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed Technical Specification revision does not involve an SHC because the revision would not:

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

The proposed TS changes are editorial in nature and do not alter or effect the design, operation, maintenance[,] or surveillance associated with MP–3 [Millstone Nuclear Power Station, Unit No. 3] [s]tructures, [s]ystems, and [c]omponents (SSC) during normal or accident operations. Since the SS[Cs] are not altered[,] the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed TS changes are editorial in nature and do not alter or effect the design, operation, maintenance[,] or surveillance associated with MP–3 [s]tructures, [s]ystems, and [c]omponents (SSC) during normal or accident operations. Since the Units SS[Cs] have not been modified physically, or operationally[,] due to procedure changes prompted by this TSCR [Technical Specification Change Request], the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

These proposed TS changes are editorial and do not impact any MP–3 design or operational requirements. MP–3 system performance and operating limits are not affected; therefore[,] the proposed change does not involve a significant reduction in the margin of safety.

In conclusion, based on the information provided, it is determined [by NNECO] that the proposed revision does not involve a[n] SHC.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 22, 1999.

Description of amendment request: The Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) contained in Appendix A to the Operating Licenses would be amended to eliminate a surveillance requirement for the Reactor Recirculation System. This proposed TS change request involves revising the TS to delete Surveillance Requirement 4.4.1.1.2, and associated TS Administrative Controls Section 6.9.1.9.h, which requires that each Reactor Recirculation System pump motor generator (MG) set scoop tube mechanical and electrical stop be demonstrated OPERABLE with the overspeed setpoints less than or equal to the setpoints as noted 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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed TS changes do not make any physical changes to the fuel, or the way the fuel responds to a transient or accident. The radiological barriers are not compromised. The fuel will continue to be operated to analyzed operating limits. No new failure mode is introduced.

Prior to the removal of the Recirculation System Master Flow Controller at LGS, the bounding postulated event involving an increase in reactor coolant system flow rate was the dual pump slow flow runout event not terminated by SCRAM. The requirements surrounding the MG set stops were established to mitigate consequences during a dual pump slow flow runout by providing a limit on the maximum core flow. The MG set stop requirements were not established to prevent an accident. The potential common mode failure required for a dual pump slow flow runout event was eliminated with the removal of the Master Flow Controller. The elimination of the Master Flow Controller does not increase the probability of other core flow increase events, or of any other events previously analyzed.

Revised generic flow biased ARTS [APRM (average power range monitor)/ RBM (rod block monitor) Technical Specifications Improvement] thermal limits that do not take credit for MG set stops have been developed for LGS, Units 1 and 2. Adherence to approved flow biased ARTS thermal limits identified in the LGS, Units 1 and 2, Core Operating Limits Reports (COLRs) ensure that fuel design limits are not exceeded. Maintaining fuel design limits results in no change in the consequences of accidents previously evaluated.

The single pump slow flow runout does not terminate by Main Steam Isolation Valve (MSIV) closure or generator load reject. As a result, the single pump runout event does not result in any significant pressurization and does not represent a challenge to the reactor coolant pressure boundary. MSIV closure with associated SCRAM on high neutron flux, as confirmed in the cycle specific Supplemental Reload Licensing Report (SRLR), remains the bounding reactor pressure vessel overpressurization event for LGS, Units 1 and 2. In addition, there are no other associated impacts to the plant resulting from a single pump runout. Therefore, the integrity of radiological barriers will not be compromised.

Although there is no longer a safety need to demonstrate operability of the MG set stops, there still is an operational need to have the MG set stops for the Reactor Recirculation System (RS). Damage to the jet pump sensing lines could occur if the resonance frequency of the sensing lines is reached. Jet pump sensing line tests established a conservative pump speed limit (1650 rpm for Unit 1, no limit for Unit 2) to preclude sensing line resonance. The MG set stop setpoint bounded the operationally required setpoint. The operationally required MG set stop setpoint to preclude jet pump sensing line resonance will continue to be controlled administratively via approved plant procedures. The proposed TS changes do not adversely impact the RS, or introduce new or unanalyzed operating conditions for the RS. The MG sets will not exceed their previously analyzed maximum 57.5 Hz with the stops removed.

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 do not make any physical changes to the fuel, or the way the fuel responds to a transient or accident. The radiological barriers are not compromised. The fuel will continue to be operated to analyzed operating limits. No new failure mode is introduced.

The proposed TS changes do not create new operating conditions that have not been evaluated. Removal of the Recirculation Master Flow Controller eliminates the possibility of a single failure initiated common mode event. Since the possibility of a common failure has been eliminated, the most limiting recirculation runout event is a one pump slow flow runout. This is the same kind of postulated accident as that previously evaluated, only it involves one pump instead of both pumps. 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 do not make any physical changes to the fuel, or the way the fuel responds to a transient or accident. The radiological barriers are not compromised. The fuel will continue to be operated to analyzed operating limits. No new failure mode is introduced.

Single pump runout based, generic flow biased ARTS thermal limits that do not take credit for MG set stops have been developed for LGS, Units 1 and 2. Adherence to approved ARTS-based flow biased thermal limits identified in the LGS, Units 1 and 2, COLRs and implemented in the plant process computer are sufficient to maintain the margin of safety as delineated in TS Sections 3/4.2.1, 3/4.2.3, and 3/4.2.4.

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

Based on the above review, the NRC staff concludes that 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 Section Chief: James W. Clifford.

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

Date of amendment request: January 29, 1998.

Description of amendment request: The amendment would delete the requirements for a security plan from the 10 CFR Part 50 license and technical specifications after the spent nuclear fuel is transferred to a Part 72 licensed independent spent fuel storage installation (ISFSI). Security requirements for the ISFSI would be in accordance with 10 CFR Part 72, Subpart H.

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 physical structures, systems and components of the Trojan Nuclear Plant and the operating procedures for their use are unaffected by the proposed change. The proposed elimination of the security requirements for the 10 CFR Part 50 license, is predicated on approval of the Trojan ISFSI Security Plan (PGE 1073) which will be coincident with issuance of a 10 CFR Part 72 license and upon completion of the transfer of all nuclear fuel from the spent fuel pool to the ISFSI. The planned 10 CFR 72 licensing controls for the ISFSI will provide adequate confidence that personnel and equipment can perform satisfactorily for normal operations of the ISFSI and respond adequately to abnormal events/accidents. The proposed Trojan ISFSI Security Plan (PGE 1073) will also provide confidence that security personnel and safeguards systems will perform satisfactorily to ensure adequate protection for the storage of spent nuclear fuel. Therefore, the proposed 10 CFR Part 50 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is security related, and as such, has no direct impact on plant equipment or the procedures for operating plant equipment and, therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated. Because the proposed ISFSI area will be segregated from the 10 CFR Part 50 licensed area, licensed security activities under the 10 CFR Part 50 license will no longer be necessary after all the nuclear fuel has been moved. The planned 10 CFR 72 licensing controls for the ISFSI area will provide adequate confidence that personnel and equipment can perform satisfactorily for normal operations of the ISFSI and respond adequately to normal events/accidents. Moreover, the ISFSI will be physically separate from the Trojan Nuclear Plant structures and equipment. Therefore, the proposed 10 CFR Part 50 license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The assumptions for a fuel handling and other accidents are not affected by the proposed license amendment. Because the proposed ISFSI area (that will contain the nuclear fuel) will be segregated from the 10 CFR Part 50 licensed area, licensed security activities under the 10 CFR Part 50 license will no longer be necessary. The planned 10 CFR 72 licensing controls for the ISFSI area will provide adequate confidence that personnel and equipment can perform satisfactorily for normal operations of the ISFSI and respond adequately to abnormal events/accidents. Also, the ISFSI will be physically separate from the Trojan Nuclear Plant structures and equipment. Therefore, the proposed 10 CFR Part 50 license amendment does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

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

NRC Section Chief: Michael T. Masnik.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: August 19, 1999. The August 19, 1999, submittal supersedes the February 18, 1999, submittal in its entirety (64 FR 14284).

Description of amendment request: The proposed amendment would revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) to incorporate the new Pressure/Temperature (P–T) Limits Curves consistent with the analysis results of reactor vessel specimen W. These figures are contained in Section 3/4.4.9 and are presented as Figures 3.4–2 and 3.4–3. These figures were developed using the methodology included in WCAP 14040–NP–A, ''Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'' as well as Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for Section XI, Division I.'' A reduced flange temperature requirement was included in the development of the curves, with justification provided in WCAP 15102, Revision 1, ''V. C. Summer Unit I Heatup and Cooldown Limit Curves for Normal Operation.'' Additionally, the Bases section for the Pressure/Temperature Limits would be revised to accurately reflect current industry standards and regulations. A significant portion of this Bases section would be deleted due to the information also being located in WCAP 15102, Revision 1.

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

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

The proposed changes revise the Pressure/Temperature Limits Curves to provide curves that reflect the results of the analysis performed on reactor vessel surveillance specimen W. This analysis was performed using NRC approved methodology as documented in WCAP 14040–NP–A, utilizing the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G requirements, along with ASME Code Case N–640. These curves provide the limits for operation of the Reactor Coolant System during heat up, cool down, criticality, and hydrotesting. These curves are provided without instrument uncertainties included, however, the uncertainties are included in the curves provided in the operating procedures. The limits protect the reactor vessel from brittle fracture by separating the region of acceptable operation from the region where brittle fracture is postulated to occur. Failure of the reactor vessel is not a VCSNS design basis accident, and, in general, reactor vessel failure has a low probability of occurrence and is not considered in the safety analysis. Therefore, the 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 changes revise the Pressure/Temperature Limits Curves, Section 3/4.4.9, to incorporate the results of the analysis performed on reactor vessel specimen W. There are no plant design changes or significant changes in any operating procedures. This change adjusts the heatup and cooldown curves to reflect the shift in nil-ductility reference temperature of the reactor vessel as a result of neutron embrittlement, and alternate methodology utilized to generate the curves. Therefore, the 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 margin of safety?

The proposed changes revise the Pressure/Temperature Limits Curves, Section 3/4.4.9, to incorporate the results of the analysis performed on reactor vessel specimen W. The new PT curves ensure that the 10 CFR 50 Appendix G, requirements are not exceeded during normal operation including Reactor Coolant System transients during heat up, cool down, criticality, and hydrotesting. The new PT curves were prepared, using accepted industry methodology, for a projected reactor vessel neutron exposure of 32 EFPY [Effective Full Power Years].

The new curves will serve as the basis for operating limitations, to provide margin against non-ductile fractures. The uncertainties introduced by instrumentation, forced flow and elevation differences are not reflected in the TS curves. These uncertainties will be factored into the curves presented in the operating procedures. Since administrative limits remain in place to ensure that 10 CFR 50 Appendix G limits are not challenged, the margin of safety described in the TS Bases is not reduced by the proposed change. Therefore, the change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment requests: August 11, 1999 (PCN–488).

Description of amendment requests: The proposed amendments would modify the Technical Specifications for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 to revise Surveillance Requirement (SR) 3.3.7.3 by providing allowable values in place of analytical limits for certain degraded voltage parameters, and by deleting unnecessary parameter limits in cases where plant safety is not affected. The proposed change would also delete redundant SR 3.3.7.4.

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

No.

Proposed Change Number (PCN)–488 revises the Technical Specification (TS) Surveillance Requirement (SR) acceptance criteria of the Loss of Voltage Signal (LOVS), Degraded Grid Voltage with Safety Injection Actuation Signal (DGVSS), and Sustained Degraded Voltage Signal (SDVS) relay circuits. These circuits are not accident initiators.

PCN–488 revises the TS SR acceptance requirements to make them more limiting than the present requirements. Because the revised acceptance criteria are more limiting than the present requirements, the consequences of accidents analyzed in the Updated Final Safety Analysis Report (UFSAR) are not increased. PCN–488 also revises the TS SR acceptance requirements to delete upper and lower bounds in cases where the deleted bound provides no safety benefit. Deleting bounds having no safety significance does not involve a significant increase in the probability or consequences of an accident previously evaluated.

PCN–488 deletes redundant SR 3.3.7.4, which is not in NUREG–1432, Standard Technical Specifications, Combustion Engineering Plants. Deleting a redundant requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Consequently, the proposed amendment does not result in an increase in the probability of accidents evaluated in the UFSAR.

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

No.

PCN–488 revises the TS SR acceptance criteria of the LOVS, DGVSS, and SDVS relay circuits, which are not accident initiators, and deletes a redundant SR. PCN–488 does not introduce any revision in the hardware configuration of the protective circuitry for LOVS, DGVSS or SDVS. The measurement required by the deleted, redundant surveillance is required elsewhere in the TS. For these reasons, PCN–488 does not create the possibility of any new or different kind of accident from any previously evaluated.'

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

No.

PCN–488 provides allowable values for the acceptance criteria for the TS SR for LOVS, DGVSS and SDVS. As such, the revised values are more limiting than the current values, which represent design limits. Therefore, PCN–488 does not involve a significant reduction in a margin of safety.

PCN–488 also revises the TS SR acceptance requirements to delete upper and lower bounds in cases where the deleted bound provides no safety benefit. Deleting bounds having no safety significance does not involve a significant reduction in a margin of safety.

PCN–488 additionally deletes a redundant SR. Because the deleted surveillance is required elsewhere in the TS, this action does not involve a significant reduction in a margin of safety.

For these reasons, PCN–488 does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

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: August 31, 1998, as supplemented by letters dated April 19 and August 18, 1999. The August 31, 1998, application was originally noticed in the **Federal Register** on October 21, 1998 (63 FR 56260).

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.4.9.3 by revising the cold overpressure mitigation curve to accommodate the replacement steam generators and by adding two surveillances (for the centrifugal charging pumps and the emergency core cooling system accumulators) to ensure the operability of the cold overpressure mitigation system.

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

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

Reanalysis of STP [South Texas Project, Units 1 and 2] COMS [cold overpressure mitigation system] transients to consider design characteristics of Delta-94 RSGs [replacement steam generators] has shown that maximum allowable PORV [power-operated relief valve] setpoints decrease slightly, and continue to provide design basis low temperature overpressure protection with Delta-94 steam generators. This change request incorporates the new COMS curves into Technical Specification 3.4.9.3 (Figure 3.4–4). Maximum allowable PORV setpoints decrease with Delta-94 steam generators, and are conservative compared to Model E steam generator curves. Use of the new curves with either Model E or Delta-94 steam generators conforms to the STP design basis.

These changes are based on a reanalysis that accounts for Model Delta-94 design, a decision to make calculation[s] of COMS maximum allowable PORV setpoint consistent with current industry standards as represented by WCAP–14040, and addition of two surveillances to the Technical Specification to ensure operability of COMS. Moving maximum allowable PORV setpoints in the conservative direction and adding surveillances to reinforce standard operating practice have no adverse effect on the probability or consequences of an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed PORV maximum allowable setpoint changes do not create any new operating conditions or modes, and the added surveillances have no effect except to ensure operation of COMS as designed. The slight change to the maximum allowable PORV setpoint curves for the Cold Overpressure Mitigation System accommodates Delta-94 steam generator design characteristics, and COMS continues to perform in accordance with existing requirements, which are sufficient to ensure plant safety is preserved.

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

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

The proposed change reflects design characteristics of the new Delta-94 steam generators. The change to the COMS curves is in the conservative direction and does not affect any design failure point or system limitation. Therefore, the change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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, Texas 77488.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 18, 1999.

Description of amendment request: The licensee proposed changing the Vermont Yankee Nuclear Power Station (VY) Technical Specifications by revising the reactor core spiral reloading pattern such that it begins around a source range monitor rather than from the center of the core. The offloading pattern would be the reverse sequence.

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

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

VY has determined that the proposed change to reload the reactor core in a spiral pattern beginning around a Source Range Monitor (SRM) does not involve a significant increase in the probability or consequences of an accident previously evaluated. The design basis accident associated with refueling is the Refueling Accident; i.e., the accidental dropping of a fuel bundle onto the top of the core. There is no assumption as to the core loading pattern in the analysis of this accident. The analyzed abnormal operational transients associated with refueling are: (1) the Control Rod Removal Error During Refueling, and (2) the Fuel Assembly Insertion Error During Refueling. There is no assumption as to the core loading pattern in the analyses of these transients. The Fuel Assembly Insertion Error During Refueling transient involves mislocated and rotated fuel assembly loading errors. However, a change in the approved core loading pattern has no impact on the probability of mislocating or rotating a bundle while following that pattern. Furthermore, the proposed change implements a core loading pattern that provides improved flux monitoring as compared to the pattern prescribed by the current Technical Specifications. When loading the core in accordance with the proposed change, the SRM indication will be indicative of the true flux of the loaded fuel, as the creation of flux traps (moderator filled cavities surrounded on all sides by fuel) is precluded.

The SRMs and the core loading pattern are not initiators of any accident previously evaluated. As such, the subject changes cannot affect the probability of an accident previously evaluated. The core loading pattern is not assumed in the mitigation of any accident. Since the proposed change provides improved flux monitoring by the SRMs, operators will have more accurate indication and SRM automatic trip functions will actuate based on a more accurate indication of flux. As such, any event mitigation function provided by the SRMs is enhanced by this change. Therefore, the associated changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. VY proposes to change the core reloading and offloading patterns to start and stop, respectively, at an

SRM versus the geometric center of the core as prescribed by current Technical Specifications. This ensures that flux monitoring instrumentation is always OPERABLE in the fueled region of the vessel. There is no separation of the monitoring device from the fuel by cavities of water as is the case with the pattern prescribed by the current Technical Specifications. As such, flux monitoring is enhanced during core reloading and offloading. This change is conservative relative to the current requirements. Therefore, no new or different kinds of accidents are created.

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

VY has determined that the proposed change does not involve a significant reduction in a margin of safety. Loading around the geometric center of the core as prescribed by the current Technical Specifications results in cells of moderator separating the fuel from the instrumentation monitoring its flux. This change requires the flux monitoring instrumentation to be in the fueled region, and, in so doing, provides for more accurate monitoring of core flux during core reloading and offloading. As such, the operators will have more accurate indication and SRM automatic trip functions will actuate when the actual flux reaches the trip setpoints. Therefore, this change will not result in a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: August 18, 1999.

Description of amendment request: The licensee proposed changing the Vermont Yankee Nuclear Power Station (VY) technical specifications (TSs) by revising the definition of the ''Surveillance Frequency'' to incorporate provisions that apply upon the discovery of a missed TS surveillance. The provisions would allow 24 hours to perform the surveillance before the applicable limiting condition for operation is entered.

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

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

This change does not result in any physical alteration of plant systems, structures or components; nor does the change modify the manner in which plant equipment will be operated or maintained. As a result, the proposed change does not affect any of the parameters or conditions that contribute to the initiation or mitigation of any accidents previously evaluated.

Surveillance frequencies are not assumed in the initiation of any analyzed event. Thus, conditions assumed in the plant accident analyses are unchanged. Furthermore, there is no relaxation of required setpoints or operating parameters.

Therefore, the probability or consequences of an accident previously evaluated are not significantly increased since the most likely outcome of performing a surveillance is that it does, in fact, demonstrate the system or component is operable. VY has, therefore, determined that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change will not modify the physical plant or the modes of plant operation. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. These changes to Technical Specifications do not create any new or different kind of accident since they do not involve any change to the plant or the manner in which it is operated.

Therefore, VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously [evaluated].

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

The proposed change does not affect design margins or assumptions used in accident analyses. The capability of safety systems to function and limiting safety system settings are similarly unaffected as a result of this change.

The increased time allowed (up to 24 hours) for the performance of a surveillance discovered to have not been performed, is acceptable based on the small probability of an event requiring the associated component. The requested allowance will provide sufficient time to perform the missed surveillance in an orderly manner. Without the 24 hour delay, it is possible that the missed surveillance would force a plant shutdown; thus, the plant could be shutting down while the missed surveillance is being performed. As a result of this delay, the potential for human error will be reduced. Consequently, there is no significant reduction in a margin of safety as overall plant safety is enhanced due to the avoidance of unnecessary plant shutdowns.

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

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

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

NRC Section Chief: James W. Clifford.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Unit Nos. 1 and 2, Louisa County, Virginia

Date of amendment request: August 4, 1999.

Description of amendment request: The proposed changes to North Anna Power Station (NAPS) Units 1 and 2 Technical Specification (TS) 4.4.1.6.1 and associated Bases will extend the drained reactor coolant loop verification time (verified as drained) from two hours to four hours prior to backfilling when returning the drained loop to service.

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

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

Administrative procedures ensure that the initiation of seal injection in order to establish a partial vacuum in an isolated and drained loop will not create the potential for an inadvertent and undetected introduction of under-borated water into an isolated loop prior to returning the isolated loop to service. Additionally, extension of the drained loop verification time from two hours to four hours prior to backfill operations will not significantly diminish confidence that the isolated and drained loop will, in fact, be drained at the time the back-fill evolution is initiated. Therefore, there is no measurable increase in the probability or consequences of any accident previously evaluated.

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

There are no modifications to the plant as a result of the changes. No new accident or event initiators are created by the initiation of seal injection in order to establish a partial vacuum in an isolated and drained loop, and by the extension of the drained loop verification time requirement from two hours to four hours prior to backfill operations. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

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

The proposed changes have no effect on the safety analyses assumptions. Changes acknowledge the establishment of seal injection for the Reactor Coolant Pump in the isolated and drained loop as a prerequisite for the vacuum-assisted back-fill technique and extends the drained-loop verification time from two hours to four hours prior to backfill operations. The two hour interval was established to ensure that the drained loop is verified to be drained at a point in time sufficiently close to the initiation of the back-fill evolution such that no intervening event could occur that would render the loop no longer drained. Relaxation of the drained loop verification time from two hours to four hours will not significantly diminish confidence that the isolated and drained loop will be drained at the time the back-fill evolution is initiated. Therefore, the proposed changes do not result in a reduction in a margin of safety.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch, Jr.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: April 28, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) Section 3.4.A.4 and Table 4.1–2B for Units 1 and 2. The proposed changes would reduce the minimum volume requirement for the refueling water chemical addition tank (CAT) to provide additional operating margin, and also correct administrative format errors in Table 4.1–2B.

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

Criterion 1—*Does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The probability or the consequences of an accident previously evaluated are not increased. When the revised Safety Analysis Limit minimum CAT volume of 3800 gallons was implemented, consideration was given to the effects of the proposed reduced CAT volume on containment integrity analyses, containment spray and post-LOCA sump pH analyses, and the post-LOCA recirculation switchover time interval specified in Emergency Operating Procedures. The change was determined to be acceptable as accident analyses assumptions would continue to be met. The proposed TS minimum CAT volume (3930 gallons) includes an allowance for the CAT level Channel Statistical Allowance (CSA), so that the safety analysis limit CAT volume (3800 gallons) will not be violated when the measured CAT volume (i.e., tank level) is at or above the TS minimum CAT volume limit. The proposed reduction in the TS minimum CAT volume has no bearing on the probability of occurrence of any accident previously evaluated, since neither the volume nor the sodium hydroxide inventory of the CAT have any bearing on postulated accident initiators. Furthermore, because the affected accident analyses have been evaluated and found to meet their acceptance criteria with the reduced safety analysis limit CAT volume, the consequences of an accident previously evaluated is not increased.

Criterion 2—*Does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The possibility of a new or different kind of accident than any accident previously evaluated is not created. The proposed reduction in the TS minimum CAT volume does not involve any alterations to the physical plant that would introduce any new or unique operational modes or accident precursors. Only the TS minimum CAT volume is being changed to establish an operationally feasible alarm setpoint to provide the operators additional flexibility in maintaining the required CAT volume.

Criterion 3—*Does not involve a significant reduction in a margin of safety.*

The margin of safety is not reduced. It was determined that the affected safety analyses continue to meet their respective acceptance criteria with the revised minimum CAT volume. By implementing the proposed change in the TS minimum CAT volume, a CAT level alarm setpoint may be established which includes a conservative allowance for level measurement uncertainty such that neither the proposed TS minimum CAT volume nor the Safety Analysis Limit CAT volume will be violated at the time a CAT level alarm is received. Therefore, it is concluded that the proposed change will not reduce the margin of safety.

This analysis demonstrates that the proposed amendment to the Surry Units 1 and 2 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Attorney for licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: December 29, 1998, as supplemented by letter dated July 29, 1999. The December 29, 1998, amendment application was previously noticed in the **Federal Register** on February 24, 1999 (64 FR 9023).

Description of amendment request: The amendment would revise Section 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' of the improved Technical Specifications (TSs), that were issued in Amendment 123 on March 31, 1999. The amendment would (1) add the phrase ''and Cold Overpressure Mitigation System'' to the first sentence of item 5.6.6.b that identifies the limits that can be determined by the licensee in the PTLR, and (2) replace the current list of documents listed in item 5.6.6.b by the NRC letter that will approve this amendment and the Westinghouse report, WCAP-14040-NP-A, ''Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves,'' dated January 1996. WCAP-14040-NP-A is the NRC-approved topical report that provides a methodology for developing the cold overpressure mitigation system (COMS) setpoints and RCS heatup and cooldown limit curves for Westinghouse plants, such as Wolf Creek Generating Station (WCGS).

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 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 Section Chief: Stephen Dembek.

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

The following notice was previously published as a separate individual notice. The notice content was the same as above. It was published as an individual notice either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. It is 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.

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

Date of amendment request: August 6, 1999.

Brief description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) contained in Appendix A to the Operating Licenses to incorporate a note into the TSs which will permit a one-time exemption, until September 30, 1999, from the 90°F limit stated in Surveillance Requirement (SR) 3.7.2.2. This SR currently requires that the average water temperature of the normal heat sink be less than or equal to 90°F as demonstrated on a 24-hour frequency. As stated in the proposed TS note, during the time period between approval and September 30, 1999, the average water temperature of the normal heat sink will be limited to less than or equal to 92°F.

Date of publication of individual notice in **Federal Register**: August 13, 1999 (64 FR 44243).

Expiration date of individual notice: 14 days for comments, August 27, 1999; 30 days for hearing, September 13, 1999.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: July 30, 1999.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.8, ''Ultimate Heat Sink (UHS),'' to permit a 72-hour delay in the UHS temperature restoration period prior to entering the plant shutdown required actions. This TS amendment is given as a temporary amendment change effective until September 30, 1999, after which the TS will revert back to the original TS provisions.

Date of issuance: August 24, 1999.

Effective date: August 24, 1999.

Amendment No.: 184.

Facility Operating License No. DPR–23: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 43406 dated August 10, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 8, 1999, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated August 24, 1999.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of application for amendments: March 25, 1999.

Brief description of amendments: The amendments revise various parts of the Technical Specifications (Appendix A of the Catawba operating licenses) to identify that the Trip Setpoints for the reactor trip system and engineered safety feature actuation system instrumentation are in reality Nominal Trip Setpoints.

Date of issuance: August 13, 1999.

Effective date: As of the date of issuance and shall be implemented within 45 days from the date of issuance.

Amendment Nos.: 179—Unit 1; 171—Unit 2.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24195).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 18, 1996, as supplemented December 12, 1997, February 23, June 15, and July 15, 1999; and by separate application dated October 22, 1997, as supplemented February 23, June 28, and July 15, 1999.

Brief description of amendment: This amendment implements: (1) voltage-based repair criteria for BVPS–2 steam generator tubes similar to the changes approved for BVPS–1 in License Amendment No. 198. The changes revise BVPS–2 technical specifications (TSs) 4.4.5 and 3.4.6.2 and associated Bases to reflect the guidance provided in the Nuclear Regulatory Commission's (NRC) Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' (GL 95–05). Additionally, BVPS–2 TS Table 4.4–2 is revised to reference TS 6.6 for reporting requirements. (2) reduced reactor coolant system (RCS) specific activity limits in accordance with the NRC's guidance provided in GL 95–05. The definition of Dose Equivalent I–131 is replaced with the Improved Standard TS definition in the first sentence, and an equation is added based on dose conversion derived from the International Commission on Radiation Protection (ICRP) ICRP–30. TS 3.4.8, Specific Activity, is revised by reducing the Dose Equivalent I–131 limit from 1.0 [micro] Ci [curies]/gram to 0.35 [micro] Ci [curies]/gram for the 48-hour limit and from 60 [micro] Ci [curies]/gram to 21 [micro] Ci [curies]/gram for the maximum instantaneous limit. Item 4.a in TS Table 4.4–12, Primary Coolant Specific Activity Sample and Analysis Program; TS Figure 3.4–1, and the Bases for TS 3/4.4.8 are also modified to reflect the reduced Dose Equivalent I–131 limit.

The February 23, 1999, letter provided a revised control room dose calculation in support of both the June 18, 1996, and October 22, 1997, amendment requests. Importantly, this calculation assumed the lower allowable primary-to-secondary leak rate limit associated with the June 18, 1996, submittal, and the reduced RCS specific activity limits associated with the October 22, 1997, submittal. Because of this interdependence, the changes of the first amendment request must be implemented concurrently with those of the second in order for the supporting analysis to remain valid. Hence, both of these license amendment requests have been combined into this single amendment.

Date of issuance: August 18, 1999.

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

Amendment No: 101.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64109) and March 25, 1998 (63 FR 14485). The December 12, 1997, February 23, June 15, June 28, and July 15, 1999, letters provided additional information but did not change the initial proposed no significant hazards consideration determinations or expand the amendment requests beyond the scope of the **Federal Register** notices.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 18, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 24, 1998, as supplemented by letters dated February 25 and July 14, 1999.

Brief description of amendments: The amendments revise the administrative sections of the Technical Specifications to reflect the approved consolidated quality assurance program, clarify the responsibilities of the shift technical advisor position on shift, simplify the contents of the monthly operating report description, complete the relocation of the fire protection requirements from the Technical Specifications, and replace selected position titles with descriptions of functional responsibility.

Date of issuance: August 26, 1999.

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

Amendment Nos.: 198 and 209.

Facility Operating License Nos. DPR–51 and NPF–6: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 27, 1999 (64 FR 4156).

The February 25 and July 14, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: November 22, 1998.

Brief description of amendment: This amendment revises the reactor thermal margin safety limit lines and flow rates stated in the St. Lucie, Unit 1, technical specifications (TS). The amendment also updates the reference for dose conversion factors used in Dose Equivalent Iodine-131 calculations, makes administrative changes to the criticality analysis uncertainty described in TS 5.6.1.a.1, updates the analytical methods used in determining core operating limits listed in TS 6.9.1.11, and revises the TS Bases for the steam generator pressure-low trip setpoint.

Date of Issuance: August 18, 1999.

Effective Date: August 18, 1999.

Amendment No.: 163.

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

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6696).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 18, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

Date of application for amendment: October 30, 1998, as supplemented December 31, 1998, and May 12, 1999.

Brief description of amendment: The amendment approves changes to the Improved Technical Specifications to reflect the use of Topical Report BAW–2421 for fluence determination and changes to the low temperature over-pressure protection limits. Changes to the CR–3 Pressure/Temperature Limits Report to reflect plant operation to 32 Effective Full Power Years were included in the submittal.

Date of issuance: August 12, 1999.

Effective date: As of date of issuance, to be implemented prior to commencing Cycle 12 operation.

Amendment No.: 183.

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

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71965). The supplemental letters dated December 31, 1998, and May 12, 1999, did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 30, 1998.

Brief description of amendment: The Amendment revises Technical Specifications (TS) to allow both doors of the containment personnel air lock to be open during fuel movement and adds a provision for an outage equipment hatch.

Date of issuance: August 16, 1999.

Effective date: August 16, 1999.

Amendment No.: 184.

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

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 30, 1998, as supplemented April 8 and July 8, 1999.

Brief description of amendment: Revises Technical Specifications for the Control Room Emergency Ventilation System and the Ventilation Filter Test Program.

Date of issuance: August 23, 1999.

Effective date: August 23, 1999.

Amendment No.: 185.

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

Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64115). The April 8 and July 8, 1999, supplements did not change the original proposed no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1999.

No significant hazards consideration comments received: No.

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

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: December 3, 1998, as supplemented by letters dated March 26, April 16, May 7, May 21, June 4, June 15, and June 29, 1999.

Brief description of amendment: The amendment revises the Technical Specification Figure 2.1–1 ''Core Protection Safety Limit,'' and Figure 2.1–3 ''Core Protection Safety Bases'' to reflect a decrease in reactor coolant system flow resulting from a revised analysis to allow operation of the TMI–1 facility with an average of 20 percent of the steam generator tubes plugged, and no more than 25 percent plugged in either generator.

Date of issuance: August 19, 1999.

Effective date: As of the date of demonstration of a satisfactory emergency feedwater pump flow test, as described in the license amendment and documented by the licensee, to be performed during the 13R refueling outage scheduled to begin September 10, 1999, and shall be implemented within 30 days of that date.

Amendment No.: 214.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 30, 1998 (63 FR 71967). The supplements dated March 26, April 16, May 7, May 21, June 4, June 15, and June 29, 1999, are within the scope of the original notice and do not change the proposed no significant hazards consideration finding.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: February 2, 1999 as supplemented July 29, 1999.

Brief description of amendment: The amendment expands the scope of systems and test requirements for post-accident reactor building sump recirculation engineered safeguards features systems and increases the maximum allowable leakage of TS 4.5.4 from 0.6 gallons per hour (gph) to 15.0 gph.

Date of issuance: August 24, 1999.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 215.

Facility Operating License No. DPR–50. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14283).

The supplemental letter did not change the initial no significant hazards consideration determination or the **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendment: March 5, 1999.

Brief description of amendment: The amendments relocate certain Technical Specifications (TSs) Section 6.0 administrative controls to the NRC-approved Northeast Utilities Quality Assurance Program (NUQAP) Topical Report. Specifically, Sections 6.2.3 (Unit 3 only), 6.5, 6.6 (partial), 6.7 (partial), and 6.10. The amendments also delete parts of Section 6.6 and 6.7 because their requirements are duplicated in existing regulations or elsewhere in the TSs. In addition, the amendments modify the table of contents and other TS sections to incorporate the aforementioned changes (e.g., correct references).

Date of issuance: August 13, 1999.

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

Amendment Nos.: 239 and 173.

Facility Operating License Nos. DPR–65 and NPF–49: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 7, 1999 (64 FR 17027).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: 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.

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

Date of application for amendments: February 12, 1999, as supplemented July 8, 1999. The July 8, 1999, letter provided clarifying information and did not change the original no significant hazards consideration determination.

Brief description of amendments: Administrative changes to correct typographical and editorial errors in Technical Specifications introduced in previous amendments.

Date of issuance: August 23, 1999.

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

Amendments Nos.: 228 and 231.

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24200).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 20, 1998, as supplemented by letter dated June 25, 1998.

Brief description of amendments: These amendments modified technical specification surveillance requirement, 3.8.1.4, to allow increases in the minimum fuel oil required to be stored in the day tanks for emergency diesel generators.

Date of issuance: August 23, 1999.

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

Amendment Nos.: 185 and 159.

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

Date of initial notice in **Federal Register**: January 27, 1999 (64 FR 4159).

The supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 19, 1997, as supplemented June 1, 1998, and May 13, 1999.

Brief description of amendments: The amendments revise TS 3.4.9, Pressurizer, to reduce the allowable pressurizer water volume for pressurizer operability. The allowable water volume is also revised to a percent pressurizer level of 57 percent.

Date of issuance: August 19, 1999.

Effective date: August 19, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—155; Unit 3—146.

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

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14488).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 19, 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 California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of application for amendments: September 4, 1998, as supplemented December 8, 1998, and February 16, 1999 (PCN 493).

Brief description of amendments: The amendments revise Technical Specification 3.4.10, Pressurizer Safety Valves, to increase the as-found pressurizer safety valve setpoint tolerances.

Date of issuance: August 19, 1999.

Effective date: August 19, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—156; Unit 3—147.

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

Date of initial notice in **Federal Register**: February 10, 1999 (64 FR 6711). The licensee's letters dated December 8, 1998, and February 16, 1999, provided clarifications and additional information that were within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 19, 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., Docket No. 50–348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama.

Date of amendment request: April 23, 1999, as supplemented by letters dated July 22, July 30 and August 12, 1999.

Brief Description of amendment: The amendment adds an additional condition to the license which allows Southern Nuclear Operating Company to operate Unit 1 for Cycle 16 based on a risk-informed approach to evaluate steam generator tube structural integrity.

Date of issuance: August 17, 1999.

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

Amendment No.: 143.

Facility Operating License No. NPF–2: Amendment revises the Facility Operating License to add a license condition.

Date of initial notice in **Federal Register**: June 16, 1999 (64 FR 32291).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 17, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 22, 1999, as supplemented July 15, 1999.

Brief description of amendments: The amendments revised Technical Specification 3/4.7.1.6, ''Atmospheric Steam Relief Valves,'' and added a new Technical Specification for atmospheric steam relief valve instrumentation, to ensure that the automatic feature of the steam generator power-operated relief valves (i.e., the atmospheric steam relief valves) remains operable during Modes 1 and 2.

Date of issuance: August 19, 1999.

Effective date: August 19, 1999, to be implemented within 30 days.

Amendment Nos.: Unit 1—114; Unit 2—102.

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

Date of initial notice in **Federal Register**: April 21, 1999 (64 FR 19565).

The July 15, 1999, supplement provided revised Technical Specification pages and clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 3, 1999 (TS 397).

Brief description of amendment: The Amendments change the Technical Specifications (TS) by reducing the Allowable Value used for Reactor Vessel Water Level—Low, Level 3 for several instrument functions.

Date of issuance: August 16, 1999.

Effective date: August 16, 1999.

Amendment Nos.: 260 and 219.

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

Date of initial notice in **Federal Register:** July 14, 1999 (64 FR 38037).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 16, 1999, as supplemented June 9, 1999.

Brief description of amendment: The amendment clarifies the inservice inspection requirements regarding the granting of relief from the American Society of Mechanical Engineers (ASME) Code requirements by the NRC. The amendment also made changes to reflect previous NRC approval of the use of ASME Code Case N–560.

Date of Issuance: August 13, 1999.

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

Amendment No.: 172.

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

Date of initial notice in **Federal Register:** July 14, 1999 (64 FR 38037).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 24, 1999.

Brief description of amendment: The amendment clarifies the basis for the reactor protection system bypass of the turbine stop valve closure and turbine control valve fast closure scram signals at low power. The amendment clarifies that the analytical basis for this bypass corresponds to a fraction of reactor rated thermal power and not other measures of power, for instance, turbine power.

Date of Issuance: August 13, 1999.

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

Amendment No.: 173.

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

Date of initial notice in **Federal Register:** July 14, 1999 (64 FR 38038).

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

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 1st day of September 1999.

For the Nuclear Regulatory Commission.

Suzanne C. Black,

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

[FR Doc. 99–23300 Filed 9–7–99; 8:45 am]

BILLING CODE 7590–01–P

POSTAL RATE COMMISSION

Privacy Act of 1974; System of Records

AGENCY: Postal Rate Commission.

ACTION: Notice of alterations to Privacy Act system of records.

SUMMARY: The Commission proposes to amend its system of records.

The proposed changes will update the system and ensure consistency with the Privacy Act of 1974, as amended.

DATES: See **SUPPLEMENTARY INFORMATION** section.

ADDRESSES: Send comments to the attention of Margaret P. Crenshaw, Secretary, Postal Rate Commission, 1333 H Street NW., Washington, DC 20268–0001.

FOR FURTHER INFORMATION CONTACT: Stephen L. Sharfman, General Counsel, Postal Rate Commission, at 202–789–6820.

SUPPLEMENTARY INFORMATION: The Postal Rate Commission gives notice, in accordance with the Privacy Act of 1974, 5 U.S.C. 552a(e)(4), of its systems of records and their routine uses, which have changed since the Commission's last publication of a notice. The Commission is also proposing revisions in its rules implementing the Privacy Act, contained in 39 CFR part 3003, to clarify their application and to shorten and simplify their language. These changes will also be published in the **Federal Register**.

PRC–1. To date, the Commission's sole system of records for Privacy Act purposes has been PRC–1, named *Official Personnel Files.* This system consists of information pertaining to Commission personnel generally. However, it does not explicitly include all related records maintained by the Commission, such as information regarding travel by Commission personnel on official business. In order to indicate clearly that all such information is included in the system, the Commission is replacing the previously-described PRC–1 with a more comprehensive system extending to all personnel, pay, leave and travel records. This new system, to be named *Personnel, Pay, Leave,* and Travel, will continue to be designated PRC–1. This system is described in the first section of Appendix A to Order No. 1256.

The Commission is also revising its statement of the routine uses of records contained in PRC–1. Two previously published routine uses are being abolished because they have not occurred in actual practice, and thus are apparently unnecessary. Other routine uses have been reworded, either to accommodate expansions in the use of records made by the Commission or the Postal Service, or to conform with language recommended by the Office of Management and Budget (OMB). The two pre-existing routine uses that encompass litigation-related disclosures have been combined into a single category.

The system notice also contains new routine uses either required by law or which the Commission anticipates may be necessary in the performance of agency business. These include disclosure of information to the National Archives and Records Administration, to agency contractors, and to the OMB for potential private relief legislation. One of these new routine uses reflects the requirement that federal agencies report wage information quarterly to the Parent Locator Service, as prescribed by Pub. L. 104–193, the Personal Responsibility and Work Opportunity Reconciliation Act.

The system notice does not contain a routine use for any computer matching activities that might be performed on records contained in PRC–1, as the Commission has not performed such matching activities in the past, and does not intend to do so in the future. However, the Commission provides payroll records to the Postal Service for routine processing, and it is possible that the Postal Service might use information about Commission personnel in a computer matching activity. In order to fulfill its statutory obligations regarding potential matching activities, particularly under the Computer Matching and Privacy Protection Amendments of 1998 (Pub.L. 100–508), the Commission is transmitting a notice informing the Postal Service of its policy that use of employee records for computer matching may be conducted only with express Commission approval, and requesting the Postal Service to exclude Commission employees from any matching activities it otherwise conducts.

PRC–2. As noted above, the revised PRC–1 will incorporate all Commission records pertaining to its employees. Virtually all other information in the Commission's possession concerning individuals occurs in the pleadings and other filings submitted by participants in the Commission's postal rate, mail classification, and other official public proceedings. Note: The Commission maintains a short press list containing the names, affiliations, addresses, and telephone numbers of reporters in their professional capacity. In the Commission's view, this list does not qualify as a system of records for Privacy Act purposes. Various Commission offices also maintain correspondence files that may contain some information about individuals in