inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. Publically available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 6th day of June 2000.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 00–15002 Filed 6–13–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meetings

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of June 12, 19, 26, July 3, 10, and 17, 2000.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of June 12

Tuesday, June 13, 2000

9:25 a.m.
Affirmation Session (Public Meeting)
a: Final Rule—Clarification of Regulations to Explicitly Limit Which Types of Applications Must Include Antitrust Information

9:30 a.m.
Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: Paul Lohaus, 301–415–3340)

1:00 p.m.
Meeting with Korean Peninsula Energy Development Organization (KEDO) and State Department (Public Meeting) (Contact: Donna Chaney, 301–415–2644)

Week of June 19—Tentative

Tuesday, June 20, 2000

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

9:30 a.m.
Briefing on Final Rule—Part 70—Regulating Fuel Cycle Facilities (Public Meeting) (Contact: Theodore Sherr, 301–415–7218)

1:30 p.m.
Briefing on Risk-Informed Part 50, Option 3 (Public Meeting) (Contact: Mary Drouin, 301–415–6675)

Wednesday, June 21, 2000

10:30 a.m.
All Employees Meeting (Public Meeting) (''The Green'' Plaza Area)

1:30 p.m.
All Employees Meeting (Public Meeting) (''The Green'' Plaza Area)

Week of June 26—Tentative

There are no meetings scheduled for the Week of June 26.

Week of July 3—Tentative

There are no meetings scheduled for the Week of July 3.

Week of July 10—Tentative

Tuesday, July 11

9:25 a.m.
Afirmation Session (Public Meeting) (If necessary.)

Week of July 17—Tentative

There are no meetings scheduled for the Week of July 17.

*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: June 9, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–15159 Filed 6–12–00; 1:31 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 May 20, 2000, through June 2, 2000. The last biweekly notice was published on May 31, 2000.

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

By July 14, 2000, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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: Docketing and Services Branch, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendment request: April 25, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.9.5, ''Communications'' to allow movement of a control rod in a fueled core cell in Operational Condition 5, to be exempt from the communication requirements of TS Section 3/4.9.5 when the control rod is moved with its normal drive 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:

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

TS Section 3/4.9.5 requires that direct communications be maintained between the control room and the refueling platform personnel during Core Alterations in Operational Condition 5. The requirement to have direct communications maintained between the control room and the refueling platform personnel does not have an effect on any accident previously evaluated or the associated accident assumptions. Thus, the proposed changes do not significantly increase the probability of an accident previously evaluated.

The proposed changes do not adversely effect the integrity of the reactor coolant system or secondary containment. As such, the radiological consequences of previously evaluated accidents are not changed. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

The proposed changes do not affect the assumed accident performance of any structure, system, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

LaSalle County Station, Units 1 and 2, exercise control rods during Core Alterations in Operational Condition 5. The required plant conditions for this control rod movement are specified in TS Section 3/4.9.3, ''Control Rod Position.'' TS Section 3/4.9.3 allows the movement of one control rod at a time, in a fueled core cell, under control of the reactor mode switch Refuel position one-rod-out interlock. The exercising of control rods under the control of the reactor mode switch Refuel position one-rod-out interlock is controlled by operators in the control room and does not occur when fuel is being moved in the reactor pressure vessel (RPV).

The proposed changes do not affect the margin of safety as the movement of a control rod will continue to satisfy the requirements of TS Section 3/4.9.3 and will not occur when fuel is being moved in the RPV.

Thus, this proposed change does not involve a significant reduction in a margin of safety.

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

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: April 28, 2000.

Description of amendment request: The proposed amendments would revise License Condition 2.C.(37) for Unit 1 and License Condition 2.C.(21) for Unit 2, to specify the types of fuel movements that cannot be performed during refueling unless all control rods are fully inserted.

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

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

The proposed changes to LaSalle County Station, Unit 1, License Condition 2.C.(37) and Unit 2 License Condition 2.C.(21), will require that control rods be fully inserted during the loading and shuffling of fuel assemblies during refueling in Operation Condition 5. The requirement to have control rods fully inserted during the loading or shuffling of fuel assemblies, during a refueling in Operational Condition 5, does not have an effect on any accident previously evaluated. The removal of fuel assemblies from the RPV does not affect the initiators or assumptions of a previously analyzed accident, including inadvertent criticality. Thus, the probability of the occurrence of an accident previously evaluated is not increased.

The proposed changes do not affect the analyzed refueling accidents, the integrity of the Reactor Coolant System or Secondary Containment. Thus, the radiological consequences of an accident previously evaluated are not increased.

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

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

The proposed changes to the Unit 1 and 2 License Conditions do not affect the assumed accident performance of any structure, system, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

The shutdown margin required during a refueling [outage] is specified in Technical Specifications (TS) Section 3/4.1.1, ''Shutdown Margin.'' The required shutdown margin ensures that the core will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. The single failure inadvertent criticality concerns, during a refueling, are an unexpected withdrawal of a control rod and the loading of a fuel assembly into the wrong core cell location. The analysis of these single failure inadvertent criticality concerns, for a fully loaded core, has determined that the most limiting event is the unexpected withdrawal of the highest worth control rod from a fueled cell.

The proposed changes, to the Units 1 and 2 License Conditions, will prohibit the loading and shuffling of any fuel assembly within the RPV unless all control rods are fully inserted during a refueling in Operational Condition 5. The unloading of a fuel assembly will be consistent with the fuel assembly and control rod requirements of TS Sections 3/4.9.10.1, ''Single Control Rod Removal,'' and 3/4.9.10.2, ''Multiple Control Rod Removal.'' These TS requirements ensure that the proposed changes to the license conditions will provide assurance that the current analysis for an unexpected withdrawal of the highest worth control rod from a totally fueled core remains bounding during a refueling outage.

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

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

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: May 1, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.8.1, ''A. C. Sources—Operating,'' to permit functional testing of the emergency diesel generators to be performed during power operation.

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

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

The function of the emergency diesel generators (EDGs) is to supply emergency power in the event of a loss of offsite power. Operation of the EDGs is not a precursor to any accident. Therefore, the proposed change to permit the 24-hour functional test of the EDGs to be performed during power operation does not increase the probability of an accident previously evaluated.

The EDG that is being tested will be available to supply emergency loads within the required time to mitigate an accident. In addition, the remaining required EDGs will be operable during the test. Furthermore, with any one EDG inoperable the remaining EDGs are capable of supporting the safe shutdown of the plant. Therefore, the consequences of an accident previously evaluated are not significantly changed.

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

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

The proposed changes to the 24-hour functional surveillance test will not affect the operation of any safety system or alter its response to any previously analyzed accident. The EDG will automatically transfer from the test mode of operation, if necessary, to supply emergency loads in the required time. This mode of operation is used for the monthly surveillance of the EDGs. Therefore, no new plant operating modes are introduced.

In the event the EDG fails the functional test, it will be declared inoperable and the actions required for an inoperable EDG will be performed. The remaining required EDGs will be maintained operable and are capable of feeding the loads necessary for safe shutdown of the plant. This addresses the concerns raised in the NRC Information Notice 84–69, ''Operation of Emergency Diesel Generators,'' regarding the operation of EDG[s] connected in parallel with offsite power. The Information Notice discusses EDG configurations that have the potential to lead to a complete loss of offsite and onsite power to safety buses. In summary, the proposed changes do not adversely affect the performance or the ability of the EDGs to perform their intended function.

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

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

The proposed changes will not reduce availability of the EDG being tested to provide emergency power in the event of a loss of offsite power. If a loss of offsite power with a loss of coolant accident occurs during the surveillance test, the emergency bus would de-energize and shed load. The EDG would then transfer from the test mode to the emergency mode. It would then be available to automatically supply emergency loads. In addition, the remaining required EDGs would be maintained operable during the test. Furthermore, with any one EDG inoperable, the remaining EDGs are capable of supporting the safe shutdown of the plant. The time required for the EDG being tested to pick up emergency loads will not be affected by performing the 24-hour functional test during power operation.

The proposed changes do not affect the assumptions or consequences of the analyzed accidents. Therefore, the proposed changes do not change any assumed safety margins.

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

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

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

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: April 13, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement (SR) 3.3.1.1.10 for Function 8 of Table 3.3.1.1–1 and SR 3.3.4.1.2.a. for reactor protection system (RPS) and end of cycle (EOC) recirculation pump trip instrumentation to extend the frequency of these SRs from 18 to 24 months.

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

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

Actuation of the TTV [turbine throttle valve] position switches is considered in the Turbine Trip accident analysis in Chapter 15 of the WNP–2 Final Safety Analysis Report. The valve position switches are assumed to function normally at greater than 30% reactor power level to initiate a reactor scram to mitigate pressure increase and an RPT [recirculation pump trip] to terminate jet pump flow in the accident analysis. The extension of the Channel Calibration surveillance interval to 24 months does not impact the normal function of the switches that is assumed in the accident analysis. There is no increase in probability or consequences represented by the proposed amendment.

Therefore, the extension of the surveillance intervals 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.

Historical maintenance and surveillance data indicate there is no effect on the performance of the TTV position switches resulting from an extension of the SR interval from 18 to 24 months. To ensure reliability, WNP–2 periodically replaces the TTV position switches according to the manufacturers' recommendation. The surveillance interval extension does not involve a change in design or a change of switch function. There is no increase in the probability of failure expected from the interval extension that could result in a different kind of accident from any previously evaluated.

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

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

Closure of the TTVs isolates the main turbine as a heat sink producing reactor pressure and neutron flux transients. Eight TTV limit switches (two per valve) function to actuate RPS and an EOC RPT to mitigate these transients and terminate jet pump flow. High pressure and flux transients also actuate RPS resulting in negative reactivity insertion should there be a failure of the TTV position switches. Additionally, historical maintenance and surveillance records indicate that the TTV position switches will operate within the necessary range and accuracy with the extension of the SR interval because no position adjustment has been necessary during past TTV position switch surveillance activities.

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

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

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

NRC Section Chief: Stephen Dembek

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

Date of amendment request: May 8, 2000.

Description of amendment request: The proposed amendment would change the River Bend Station, Unit 1 (River Bend or RBS), Technical Specifications (TSs) to remove the Fuel Building and the fuel building ventilation system from the requirements associated with the Secondary Containment boundary during operational Modes 1, 2, and 3.

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

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

The proposed changes to the Technical Specifications involve removing the Fuel Building and the fuel building ventilation system from the requirements associated with the Secondary Containment boundary. The changes result in conservatively assuming that all annulus bypass leakage following a DBA [design basis accident] LOCA [loss-of-coolant accident] are directed to the environment for the duration of the accident. Since the proposed changes only affect functions that are required subsequent to a LOCA or fuel handling accident (FHA), the proposed changes have no [a]ffect on the probability of an accident. The Fuel Building portion of the Secondary Containment boundary is not an active component that could affect the proper operation of any other essential safety feature or component. Removal of the Fuel Building from the Secondary Containment boundary does not affect any other safety-related system, component, or structure that would increase the probability of an accident previously evaluated. The proposed change only has an impact on the dose consequences of the design basis accident and does not have any affect on the accident precursors or other accident mitigating features.

A plant-specific radiological analysis has been performed to assess the affects of the proposed change in the annulus bypass leakage release pathway in terms of Control Room and off-site doses following a postulated design basis LOCA. The calculated doses for all offsite and onsite evaluation points are within the 10 CFR [Code of Federal Regulations] Part 100 criteria for offsite doses and within the General Design Criterion 19 of 10 CFR Part 50 for the Control Room.

The calculated offsite DBA LOCA doses due to the proposed changes result in an increase of less than 3 percent due to releasing all annulus bypass leakage directly to the environment. The control room doses exhibit the largest percentage increase in the thyroid dose due to the increase in unfiltered and untreated iodine released to the environment, the release rate to the environment, and the changes in the control room atmospheric diffusion coefficient due to dual air intakes. However, the change in control room thyroid dose reduces the margin to the regulatory limit by only 4 percent. The calculated doses for all offsite and onsite evaluation points are not significantly increased and remain within the 10 CFR Part 100 criteria for offsite doses and within the General Design Criterion 19 of 10 CFR Part 50 for control room.

The proposed changes also include relaxation of requirements for the fuel building and fuel building ventilation system except during the movement of "recently" irradiated fuel. The term "recently irradiated" is defined as "fuel that has occupied part of a critical reactor core within the previous 11 days." This change is justified based on the irradiated fuel source term decay period. River Bend currently evaluates three FHA scenarios, one for the fuel building and two for containment. The FHA–FB [Fuel Building] scenario would be impacted by the proposed changes since the scenario assumed filtration for the duration of the release. However, the proposed changes are bounding in their entirety by the FHA dose evaluation prepared in support of Amendment 85, as revised to support Amendment 110. The current analysis assumes that a FHA occurs with the containment personnel air locks (PAL) open, thus, no credit is taken for primary containment after an 11-day source term decay period. The release rate assumed in that analysis bounds the Fuel Building's normal ventilation rate by a factor of approximately 3 and easily meets Regulatory Guide 1.25 assumptions. All other data and assumptions (other than decay time of course) are identical for the two analyses and thus, the Amendment 85 analysis is valid for the Fuel Building.

It is therefore concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The operation of River Bend Station, in accordance with the proposed amendment, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes affect the TS requirements for the fuel building and fuel building ventilation system. These changes have no impact on any other safety-related system, component, or structure. The type of accident and the accident precursors are not affected by changing the annulus bypass release path. The Fuel Building portion of the Secondary Containment boundary is not an active component that could affect the proper operation of any other essential safety feature or component. Also, the accident mitigating features that are currently credited in the response to the design basis accident are unchanged by the proposed change. Changing the release path for the annulus bypass leakage does not create a new or different kind of accident from the accidents previously evaluated.

It is therefore concluded that the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

(3) The operation of River Bend Station, in accordance with the proposed amendment, does not involve a significant reduction in a margin of safety.

The fuel building and the associated fuel building ventilation filtration system are currently credited as part of the secondary containment function. The modified secondary containment boundary (excluding the fuel building) will still be capable of performing its design function of limiting offsite and control room dose to within regulatory limits. The only accident consequences that are impacted by the proposed change in the secondary containment (annulus) bypass leakage path are the dose consequences of the design basis LOCA. The previous dose analysis is changed by assuming that all annulus bypass leakage is directly to the environment instead of being released into the Fuel Building where the release would be treated by the Fuel Building Ventilation System before release. A plant-specific radiological analysis has been performed to assess the affects of the proposed change in the annulus bypass leakage release pathway in terms of Control Room and off-site doses following a postulated design basis LOCA. The proposed change required a revision to the existing LOCA dose analysis since the annulus bypass leakage release is assumed to be directly to the environment due to removal of the Fuel Building from the Secondary Containment boundary. The calculated doses for all offsite and onsite evaluation points are within the 10 CFR Part 100 criteria for offsite doses and within the General Design Criterion 19 of 10 CFR Part 50 for the Control Room.

The proposed changes to the Technical Specification requirements for the fuel building and the fuel building ventilation system when handling irradiated fuel in the fuel building are bounded by currently approved FHA analyses.

Therefore, there is no significant reduction in the margin of safety associated with postulated design basis events at RBS in allowing the proposed change to the RBS licensing basis.

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.

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 29, 1999, as supplemented by letters dated August 8, 1999, August 24, 1999, January 27, 2000, March 29, 2000, May 22, 2000, and May 31, 2000.

Description of amendment request: The proposed amendment request provides additional information to support a modification to Technical Specification (TS) 3.8.1.1 and associated Bases by extending the Emergency Diesel Generator (EDG) allowed outage time (AOT) from 72 hours to 10 days. In the supplement letter dated May 22, 2000, an alternate source for the onsite power system during the EDG maintenance outage, by way of a temporary EDG (TEDG) has been added. The application dated July 29, 1999, did not include the TEDG. This notice supercedes the biweekly **Federal Register** notice dated February 9, 2000, (65 FR 6406) based on the original application dated July 29, 1999.

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

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

Response:

The EDGs are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. As such, the EDGs are not accident initiators in any accident previously evaluated. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes to the TS will extend the allowed outage time (AOT) for a single inoperable emergency diesel generator (EDG) from the current limit of 72 hours to 10 days with the implementation of compensatory measures. These compensatory measures consist of a temporary emergency diesel generator (TEDG) capable of supplying auxiliary power to required safe shutdown loads on the EDG train removed from service. In the probabilistic risk assessment (PRA) event of a loss of offsite power, the failure of the operable EDG, and the failure of the turbine-driven emergency feedwater pump to start, the TEDG would be started and ready for load within 25 minutes. In the PRA assumptions to calculate the risk increase to core damage, 50 minutes is available until core uncovery. The AOT would be extended for: (1) preplanned maintenance work (both preventive and corrective) known to require greater than 72 hours; and (2) unplanned corrective maintenance work which may be determined to take greater than 72 hours.

The plant defense-in-depth has been preserved by the use of a TEDG to supply required safe shutdown loads. The design basis for the onsite power systems will continue to conform to 10 CFR 50, Appendix A, General Design Criterion 17.

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

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

Response:

The EDGs are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. The proposed changes to the TS will extend the allowed outage time (AOT) for a single inoperable emergency diesel generator (EDG) from the current limit of 72 hours to 10 days with the implementation of compensatory measures. These compensatory measures consist of a temporary emergency diesel generator (TEDG) capable of supplying auxiliary power to required safe shutdown loads on the EDG train removed from service. In the PRA event of a loss of offsite power, the failure of the operable EDG, and the failure of the turbine-driven emergency feedwater pump to start, the TEDG would be started and ready for load within 25 minutes. In the PRA assumptions to calculate the risk increase to core damage, 50 minutes is available until core uncovery. The AOT would be extended for: (1) preplanned maintenance work (both preventive and corrective) known to require greater than 72 hours; and (2) unplanned corrective maintenance work which may be determined to take greater than 72 hours.

The proposed change does not alter the design, configuration, and method of operation of the plant for safety-related equipment during the EDG AOT extension period. The plant defense-in-depth has been preserved by the use of a TEDG to supply power to required safe shutdown loads.

The change does involve the modification of non-safety permanent plant equipment. The modification will involve preparing a 4.16kV [kilo-volt] non-safety bus breaker for connection to the output of the TEDG. There is no change being made to the parameters within which the plant is operated, and the setpoints at which the protective or mitigative actions initiate. The design basis on which the plant was licensed will not be changed. In the PRA event of a loss of offsite power, the failure of the operable EDG, and the failure of the turbine-driven emergency feedwater pump to start, the TEDG would be started and ready for load within 25 minutes. In the PRA assumptions to calculate the risk increase to core damage, 50 minutes is available until core uncovery.

Procedures will be developed to implement onsite power system recovery action in conjunction with the present Emergency Operating Procedures (EOP) and appropriate Off Normal Procedures in the event it is necessary to use the alternate AC power source. The developed procedures support compensatory measures that provide additional assurance that if a coincident Loss of Offsite Power and failure of the operable EDG (outside the design basis of the plant) occurred during a preplanned maintenance (both preventive and corrective) or unplanned corrective maintenance extended EDG AOT outage, appropriate guidance would be available to safely shutdown the plant. There are no alterations to the existing plant procedure that will decrease assurance that the plant will remain within analyzed limits. As such, no new failure modes are being introduced that would involve any potential initiating events that would create any new or different kind of accident. The proposed change will only provide the plant some flexibility in the AOT for accomplishing preplanned maintenance (both preventive and corrective) normally performed during refueling outages and any potential unplanned corrective maintenance that may exceed the normal 72-hour AOT during plant operation in Modes 1, 2, 3, and 4. The change does not alter assumptions made in the safety analysis and licensing basis.

Therefore, since there will be no permanent hardware modifications to safety-related equipment nor alterations in the way in which the plant or equipment is operated during any design basis event, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed change does not affect the LCO's [limiting conditions for operation] or their Bases used in the deterministic analysis to establish the margin of safety. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. There is no significant impact on the margin of safety. PSA [probabilistic safety assessment] methods were used to evaluate the proposed change. The results of these evaluations indicated the risk contribution from this proposed AOT with compensatory measures implemented during this extended EDG AOT time period is small and within the Regulatory Guide 1.177 risk-informed acceptance guidelines.

Therefore, the change does not significantly impact the margin of safety, involve a permanent change in safety-related plant design, or have any affect on the plant protective barriers. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 29, 1999.

Description of amendment request: Entergy Operations, Inc. (licensee) has proposed to revise their Updated Final Safety Analysis Report (UFSAR) to discuss the probability threshold for when physical protection of safety-related components from tornado missiles is required for certain components. The proposed changes involve the use of Nuclear Regulatory Commission (NRC) approved probability risk methodology to assess the need for additional tornado missile protection and demonstrate that the probability of damage due to tornado missiles striking safety related components is acceptably low.

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

The proposed changes, *i.e.*, revising the current UFSAR descriptions addressing tornado missile barrier protection at Waterford Steam Electric Station, Unit 3 (Waterford 3) have been evaluated against these three criteria, and it has been determined that the changes do not involve a significant hazard because:

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

The associated UFSAR changes reflect use of the Electric Power Research Institute (EPRI) Topical Report, ''Tornado Missile Risk Evaluation Methodology, (EPRI NP–2005),'' Volumes 1 and 2. This methodology has been reviewed, accepted and documented in a NRC Safety Evaluation dated October 26, 1983. The NRC concluded that: ''the EPRI methodology can be utilized when assessing the need for positive tornado missile protection for specific safety-related plant features in accordance with the criteria of SRP [Standard Review Plan] Section 3.5.1.4.''

The EPRI methodology has been previously applied by other licensees to resolve tornado missile protection issues.

The results of the tornado missile hazards analysis are such that the calculated total tornado missile hazard probability for safety-related SSC's [systems, structures and components] is approximately 6.0×10^{-7} per year. This is lower than the value determined to be acceptable, i.e., 1×10^{-6} per year by the NRC Staff.

With respect to the probability of occurrence or the consequences of an accident previously analyzed in the UFSAR, the probability of a tornado reaching Waterford 3 causing damage to plant systems, structures and components is a design basis event considered in the UFSAR. The changes being proposed herein do not reduce the probability that a tornado will reach the plant. However, it was determined that there are a limited number of safety-related components that theoretically could be struck. The probability of tornado-generated missile strikes on these components were analyzed using the NRC Staff approved probability methods described above. On this basis, the proposed change is not considered to constitute a significant increase in the probability of occurrence or the consequences of an accident, due to the low probability of a tornado missile striking these components.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of previously evaluated accidents.

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

The proposed changes involve evaluation of whether any physical protection of safety-related equipment from tornado missiles is required relative to the probability of such damage without physical protection. A tornado at Waterford 3 is a design basis event considered in the UFSAR. This change involves recognition of the acceptability of performing tornado missile probability calculations in accordance with established regulatory guidance.

Therefore, the change would not contribute to the possibility of, or be the initiator for any new or different kind of accident, or to occur coincident with any of the design basis accidents in the UFSAR. The low probability threshold established for tornado missile damage to system components is consistent with these assumptions.

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

(3) The proposed activity does not involve a significant reduction on a margin of safety.

The request does not involve a significant reduction in a margin of safety. The existing licensing basis for Waterford 3 with respect to the design basis event of a tornado reaching the plant, generating missiles and directing them toward safety-related systems and components is to provide positive missile barriers for all safety-related systems and components. With the change, it will be recognized that there is an extremely low probability, below an established acceptance limit, that a limited subset of the ''important'' systems and components could be struck. The change from ''protecting all safety-related systems and components'' to ''an extremely low probability of occurrence of tornado generated missile strikes on portions of important systems and components' is not considered to constitute a significant decrease in the margin of safety due to that extremely low probability.

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: February 21, 2000.

Description of amendment request: The proposed amendment would revise the Unit 1 Updated Final Safety Analysis Report (UFSAR) descriptions for bolting material used on some Reactor Coolant System (RCS) components.

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 use of carbon steel fasteners in a borated system introduces a new failure mechanism for the fasteners, that of boric acid wastage. The materials currently specified in the [Beaver Valley Power Station] BVPS Unit 1 UFSAR are not susceptible to boric acid wastage. The probability of failure for all systems may be increased due to the additional failure mode introduced by change from corrosion resistant material to carbon steel for RCS and reactor coolant pressure boundary fasteners.

The design requirements of the [American National Standards Institute] ANSI and [American Society of Mechanical Engineers] ASME Codes are conservative in nature, in that, the stress allowable for fastener materials is less than half the yield strength of the material, thus creating a margin in the design of two or greater on structural strength. Therefore, the failure or damage of one or more non-adjacent fasteners can normally be accommodated. Additionally, the material properties (Yield and Tensile strength) of the installed (SA540 Grade B

Class 23 or 24) carbon steel fasteners are higher than that of the material identified in the UFSAR (SA453 Grade 660). It should also be noted that the use of either the carbon steel fasteners (those installed) or the stainless steel fasteners (those identified in the UFSAR) is acceptable by the design Codes (ANSI and ASME), the selection of the material for the fasteners is at the discretion of the designer and is not specified by Code requirements. When compared to carbon steel fasteners, the corrosion resistance of Grade 660 material is pertinent only if leakage is actively occurring.

The boric acid wastage concern is mitigated by the Boric Acid Corrosion Program which has systematic measures to ensure that boric acid corrosion will not lead to degradation of the reactor coolant pressure boundary. This Boric Acid Corrosion Program with its inspections provides adequate assurances that abnormal leakage will be identified and corrective actions taken prior to significant boric acid corrosion degradation of carbon steel pressure boundary components.

The NRC, in Generic Letter (GL) 88–05, recognized that boric acid solution leaking from the reactor coolant system can cause significant corrosion damage to carbon steel materials. In the GL, the NRC requested that licensees provide assurance that a boric acid monitoring program has been implemented. This program was to consist of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage or rupture. The Beaver Valley Power Station response to the GL provided assurance that a program was in place and committed to enhancements to the existing program. An NRC follow-up review was conducted and the Beaver Valley Power Station program was found to be acceptable and fulfilling the requirements of GL 88–05 (Reference: NRC Inspection Report Nos. 50–334/88–23 and 50–334/88–25)

Therefore, the proposed changes to BVPS Unit 1 UFSAR Tables 1.8–1 and 1.8–2 do not significantly increase the probability or consequences of any accident previously evaluated in the BVPS Unit 1 UFSAR.

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

During an evaluation of the fastener material to be used for the replacement of a degraded fastener, it was discovered that the BVPS Unit 1 UFSAR Tables 1.8–1 and 1.8–2 identified that corrosion resistant materials, SA453 Grade 660, were identified as being installed. The use of carbon steel fasteners in lieu of the SA453 Grade 660 fasteners identified in the UFSAR introduces the potential failure mechanism of boric acid corrosion. The corrosion damage that has occurred on MOV–RC–591 and MOV–CH–310 bolting demonstrates that corrosion damage from unchecked borated water leakage is damaging to carbon steel fasteners. Additionally, it should be noted that both of these degraded conditions were identified and repaired prior to an operational or structural concern through the application of the Boric Acid Corrosion Program.

In the design condition (non-corroded), the change to carbon steel fasteners would not affect the design basis accidents described in the UFSAR. The boric acid wastage concern is mitigated by the Boric Acid Corrosion Program which has systematic measures to ensure that boric acid corrosion will not lead to degradation of the reactor coolant pressure boundary.

In addition to the Boric Acid Corrosion Program, the body to bonnet configuration for the fasteners identified in Table 1.8–1 and 1.8–2 result in multiple fasteners for each joint. To meet the requirements of the design Codes (ANSI or ASME) for valves, the number of fasteners installed is in excess of the number of fasteners required to perform the structural function of maintaining the pressure boundary. Additionally, it is highly unlikely that all the installed fasteners would corrode in such a manner that catastrophic failure of the body to bonnet joint would result. Therefore, the multiple installed fasteners result in an installed backup to the minimum required number of fasteners necessary to maintain pressure boundary integrity.

Thus, the assumptions and consequences of the loss of pressure boundary integrity type of accident would be unchanged and would not introduce a new or different kind of accident as currently evaluated in the BVPS Unit 1 UFSAR based on the Boric Acid Corrosion Program preventing any unacceptable boric acid wastage in accordance with GL 88–05.

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

The proposed change in the Unit 1 UFSAR removing criteria requiring stainless steel fasteners for RCS and reactor coolant pressure boundary components would not involve a significant reduction in the margin of safety since current Technical Specification requirements remain unchanged and current plant programs (i.e., Boric Acid Corrosion Program inspections) provide adequate assurance from the likelihood of corroded fasteners causing an operational issue. NRC reviewed the Beaver Valley Power Station Boric Acid Corrosion Program and found the program to be acceptable to fulfill the requirements of GL 88–05 (Reference: NRC Inspection Report Nos. 50–334/88–23 and 50–334/88–25).

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.

Attorney for Licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Acting Section Chief: Marsha Gamberoni.

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

Date of amendment request: May 1, 2000.

Description of amendment request: The proposed amendment would revise the Unit 1 and 2 Technical Specification (TS) 3/4.6.4.2 Surveillance Requirement (SR). The proposed change would allow performance of the hydrogen recombiner functional test at containment pressures greater than the currently specified 13 psia. This would be accomplished by measuring the flow under normal or current test conditions (*e.g.,* atmospheric pressure) and calculating the expected system performance under design basis operating conditions. The surveillance would be revised to verify that the recombiner flow, when corrected to the post accident design conditions, is greater than or equal to the required flow. The corresponding design basis temperature for post accident recombiner operation would be included in the SR because it is required to correct the test flow to the design basis operating conditions. In order to support the calculations necessary to confirm the recombiner blower performance, the proposed change includes the addition of an equation and associated discussion to the bases. The equation will correct the measured test flow to a corresponding flow at the design basis operating pressure and temperature. In addition to the technical change described above, SR 4.6.4.2.b.3 would be modified by separating the criteria for the system blower performance and heater operation into separate parts of the same surveillance to improve the presentation of the requirements. Format and editorial changes are included as necessary to facilitate the revision of the TS text to conform to the current TS page format, and addition of text to the bases.

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 change does not result in any hardware changes to the hydrogen recombiners. Additionally, the hydrogen recombiners are not assumed to be accident initiators of any analyzed event. The proposed change revises the method for performing the hydrogen recombiner functional test specified in Technical Specification (TS) Surveillance Requirement (SR) 4.6.4.2.b.3. The proposed change to SR 4.6.4.2.b.3 does not reduce the effectiveness of the requirement and continues to verify the capability of the hydrogen recombiners to perform their design basis function consistent with the assumptions of the applicable safety analysis. Therefore, the consequences or probability of accidents previously evaluated in the UFSAR remain unchanged.

The addition of supporting TS bases text and the format and editorial changes made to the TS have no impact on plant operation or safety.

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

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does not affect any accidents previously evaluated in the UFSAR and continues to provide assurance that the hydrogen recombiners remain capable of performing their design function. The proposed change does not introduce any new failure modes or affect the probability of a malfunction.

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

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

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed change does not involve revisions to any safety limits or safety system settings that would adversely impact plant safety. In addition, the proposed change does not affect the ability of the hydrogen recombiners to perform their design function.

The proposed change revises the method for performing the hydrogen recombiner functional test specified in SR 4.6.4.2.b.3. However, the proposed change to SR 4.6.4.2.b.3 does not reduce the effectiveness of the requirement and continues to verify the capability of the hydrogen recombiners to perform their design basis function consistent with the assumptions of the applicable safety analysis.

The addition of supporting TS bases text and the format and editorial changes made to the TS have no impact on plant operation or safety.

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

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

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

NRC Acting Section Chief: Marsha Gamberoni.

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

Date of amendment request: April 27, 2000.

Description of amendment request: The proposed amendment would change the James A. FitzPatrick Nuclear Power Plant Technical Specifications by changes to the Trip Level Settings for the Residual Heat Removal (RHR) and Core Spray (CS) Pump Start Timers as well as the Automatic Depressurization System (ADS) Auto-Blowdown Timer. The amendment would also extend the Logic System Functional Test surveillance test intervals for the RHR, CS and ADS systems from 6 months to 24 months.

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

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

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

This proposed change revises the Trip Level Settings for the RHR and CS pump interlock start timers as well as the ADS auto-blowdown timers. This proposed change also extends the surveillance interval for these timers from 6-months to 24-months.

This proposed change impacts the control of systems designed to mitigate the consequences of a Loss of Coolant Accident (LOCA). These changes do not impact any of the Reactor Coolant System parameter variations listed as potential causes of threats to the fuel and Reactor Coolant Pressure Boundary listed in section 14.4.2 of the FitzPatrick UFSAR [Updated Final Safety Analysis Report] (Reference 8) [see application dated April 27, 2000]. Therefore, this proposed change does not increase the probability of an accident previously evaluated.

The changes to the control of systems designed to mitigate the consequences of postulated LOCA events are consistent with the relevant assumptions made in the FitzPatrick LOCA analysis (Reference 5) [see application dated April 27, 2000]. Therefore, the results of that analysis are not changed. Therefore, this proposed change does not increase the consequence of an accident previously evaluated. Create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposed change impacts the control of systems designed to mitigate the consequences of a Loss of Coolant Accident (LOCA). These changes do not impact any of the Reactor Coolant System parameter variations listed as potential causes of threats to the fuel and Reactor Coolant Pressure Boundary listed in section 14.4.2 of the FitzPatrick UFSAR (Reference 8) [see application dated April 27, 2000]. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Involve a significant reduction in a margin of safety.

The changes to the control of systems designed to mitigate the consequences of postulated LOCA events are consistent with the relevant assumptions made in the FitzPatrick LOCA analysis (Reference 5) [see application dated April 27, 2000]. Therefore the results of that analysis are not changed. Therefore, this proposed change does not reduce the margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni, Acting.

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

Date of amendment request: February 7, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification 4.7.1.2.b to make the surveillance requirements for Auxiliary Feedwater Pump testing consistent with that of NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.'' The Bases associated with this Technical Specification would also be revised to address the proposed changes.

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

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

The proposed changes to the Technical Specification surveillance requirements for the auxiliary feedwater pumps surveillance testing are consistent with the latest auxiliary feedwater flow hydraulic model and accident analyses. The revised minimum acceptance criteria will ensure that pump degradation, which could adversely impact the accident analyses, will be detected. The pumps will continue to operate in the same manner as assumed in the analyses to mitigate the design basis accidents.

Therefore, there will be no 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 changes to the Technical Specification surveillance requirements for the auxiliary feedwater pumps surveillance testing are consistent with the latest auxiliary feedwater flow hydraulic model and accident analyses. The proposed changes to the Technical Specification surveillance requirements and associated Bases will not affect the way the pumps are operated during normal plant operations or how the pumps will operate after an accident.

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

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

The proposed changes to the Technical Specification surveillance requirements for the auxiliary feedwater pumps surveillance testing are consistent with the latest auxiliary feedwater flow hydraulic model and accident analyses. The proposed changes to the Technical Specification surveillance requirements eliminate a potential non-conservative acceptance value and establish appropriate restrictions to ensure pump operability. The proposed change to the Technical Specifications Bases better describes the design function of the auxiliary feedwater system.

Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

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.

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

NRC Section Chief: James W. Clifford.

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

Date of application request: May 25, 2000 (ULNRC–04257).

Description of amendment request: The licensee proposed to eliminate the technical specifications (TSs) on the boron dilution mitigation system to avoid a spurious swapover event, such as occurred during the shutdown for Refueling Outage 9, about 2 years ago. This amendment would delete the limiting condition for operation, the actions, and the surveillance requirements for TS 3.3.9, ''Boron Dilution Mitigation System (BDMS),'' in the instrumentation section of the TSs for Callaway. In addition, the title of TS 3.3.9 would be removed from the Table of Contents, the Bases for the TSs would be revised, and a section on the boron dilution analysis would be added to Chapter 16 of the Callaway Final Safety Analysis Report (FSAR).

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since the associated hardware changes described in Section X of Appendix A [to the application dated May 25, 2000] do not affect any protection systems. The RTS [reactor trip system] and ESFAS [engineered safety features actuation system] instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis. The installation of an alarm on the [reactor coolant] letdown divert valve, addition of two redundant high VCT [volume control tank] water level alarms, and elimination of the automatic BDMS valve swap-over function will be performed in such a manner that all design, material, and construction standards that were applicable prior to the change are maintained.

The proposed change will modify the system interface between CVCS [chemical and volume control system] and the boron recycle system such that the RCS [reactor coolant system] and CVCS form a closed system consistent with the reanalysis assumptions. The letdown divert valve will be placed in the manual ''VCT'' mode [(1)] prior to entry into MODE 3 from MODE 2 during a plant shutdown and [(2)] prior to entry into MODE 5 from MODE 6 during a plant startup such that letdown flow is directed to the VCT, rather than to the recycle holdup tanks, except under administrative controls for planned evolutions which require a high degree of operator involvement and awareness. These administrative controls will include verification of the boron concentration of the makeup [to the reactor coolant] prior to repositioning the divert valve and restoration requirements to return the valve to the manual ''VCT'' mode upon evolution completion.

The proposed change will not affect the probability of any event initiators. The above modifications are unrelated to the initiating event for this analysis, a failure in the reactor makeup control system. The change will revise the method of detecting the event and rely on operator action for event termination. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

Since manual operator actions are being substituted for automatic actions, this amendment application was reviewed against the guidance provided in NRC Information Notice 97–78, ''Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times.'' Appendix A [to the application] demonstrates that sufficient time is available for operator action to terminate the inadvertent boron dilution event prior to criticality. Additionally, as discussed in NSAC–183, ''Risk of PWR Reactivity Accidents during Shutdown and Refueling,'' gradual inadvertent boron dilution events are not expected to cause core damage, even if they are unmitigated, due to their self-limiting nature.

The proposed change will achieve the same objective as the BDMS, i.e., the prevention of an inadvertent criticality as a result of an unintended boron dilution. The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR. Appendix A [to the application] demonstrates that sufficient time is available for operator action to terminate the inadvertent boron dilution event prior to criticality. With the reactor subcritical, there will be no increase in radiological consequences.

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

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

There are no changes in the method by which any safety-related plant system performs its safety function. The changes described in Section X of Appendix A [to the application] have no impact on any analyzed event other than inadvertent boron dilution. The physical modifications to eliminate the automatic BDMS valve swap-over function and add redundant high VCT water level alarms and a position alarm on the letdown divert valve will be implemented in accordance with existing plant design criteria. The BDMS itself has no impact on any other analyzed event. The portion of the change deleting the BDMS from the Technical Specifications, and eliminating the automatic valve swap-over function, has no other impact safety. The BDMS flux multiplication alarm will be retained as a plant design feature to provide the plant operators a diverse method for identifying a potential dilution event. Since the passive alarms to be added only provide information and do not initiate control or protection system actions, the alarms will not adversely impact other events. The position of the letdown divert valve only affects the path for letdown flow. The flow path selected for letdown does not affect any other accident analyses. Thus, the operational change to make the manual ''VCT'' mode the normal operating condition in MODES 3 through 5 has no safety impact. Procedural changes will heighten the operator awareness of potential dilution events and provide alarm response actions to mitigate potential dilution events. As such, these changes will enhance the response to inadvertent boron dilution events, but have no other safety impact. The FSAR Chapter 16 requirements for reactor coolant loop operation and high VCT water level alarm operability will also enhance the plant operators' capability to respond to an inadvertent boron dilution event. If the Chapter 16 requirements are not met, isolating the dilution source valves in MODES 3, 4, and 5 has no impact on any other accident analyses since none of the other accident analyses take credit for, or are initiated by, the flow path through these valves.

This change will affect the normal method of plant operation while in MODES 3 through 5 with regard to the control of letdown flow. In these MODES, letdown processing via the recycle holdup tanks will be allowed only under administrative controls for planned evolutions which require a high degree of operator involvement and awareness. The annunication of the letdown divert valve not being in the ''VCT'' position will further highlight system conditions to the operating staff. No other performance requirements will be affected.

In order to automatically close the VCT isolation valves, the RWST [refueling water storage tank] isolation valves must be fully open. This valve interlock feature is designed to ensure a flow path is maintained to the CCPs [component cooling pumps] during swap-over. Since the VCT isolation valves can be manually closed prior to opening the RWST isolation valves, the possibility exists for the operator to inadvertently isolate flow to the CCPs while attempting to isolate the dilution source. However, plant operating procedures provide the operators with sufficient guidance for performing a manual valve swap-over and the reanalysis demonstrates that sufficient time is available to perform the required manual actions, consistent with SRP [NRC NUREG–0800 Standard Review Plan] acceptance criteria.

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

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

The proposed change uses acceptance criteria consistent with the [NRC] Standard Review Plan, as discussed in Appendix A [to the application]. The margin of safety required of the BDMS is maintained, i.e., inadvertent boron dilution events will be terminated by timely operator actions prior to a total loss of all shutdown margin. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protective functions. There will be no impact on the overpower limit, DNBR [departure from nucleate boiling ratio] limits, F_Q, FdeltaH, LOCA PCT [loss-of-coolant accident peak cladding temperature], peak local power density, or any other margin of safety. The radiological dose consequences acceptance criteria listed in the Standard review Plan will continue to be met.

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

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 22, 2000.

Description of amendment request: The proposed amendment would remove the technical specification surveillance requirement for visual inspection of suppression chamber coating integrity once each refueling outage.

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.

The proposed change conforms the TS to current regulations, credits actions taken under GL 98–04 to address coating delamination concerns, and eliminates redundant surveillance criteria. Since reactor operation under the revised Specification is unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events or assumed mitigation of accident or transient events. The structural and functional integrity of plant systems is unaffected. Thus, there is no significant increase in the probability or consequences of accidents 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 does not affect any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created. No safety-related equipment or safety functions are altered as a result of these changes. Because it does not involve any change to the plant or the manner in which it is operated, 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 and has no effect on any initial condition. The capability of safety systems to function and limiting safety system settings are similarly unaffected as a result of this change. Thus, the margins of safety required for safety analyses are maintained.

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.

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: May 23, 2000.

Description of amendment request: This proposed change relocates those portions of Technical Specifications (TSs) related to reactor coolant conductivity and chloride requirements to the Technical Requirements Manual.

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.

The proposed change is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. Conductivity and chloride limits are not assumed to be an initiator of any analyzed event, nor are these limits assumed in the mitigation of consequences of accidents.

Therefore, 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 does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change will 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 represents the relocation of current Technical Specification requirements to the Technical Requirements Manual, based on regulatory guidance and previously approved changes for other stations. The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the Technical Requirements Manual.

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

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

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: May 23, 2000.

Description of amendment request: The proposed amendment would revise the technical specification surveillance requirements for local power range monitor calibration.

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.

The revised surveillance requirement continues to ensure that the local power range monitor (LPRM) signal is adequately calibrated. This change will not alter the basic operation of process variables, structures, systems, or components as described in the safety analyses, and no new equipment is introduced by the change in LPRM surveillance interval. Therefore, the probability of accidents previously evaluated is unchanged.

The consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident, but LPRM chamber exposure has no significant effect on the calculated thermal limits because LPRM accuracy does not significantly deviate with exposure. For the extended calibration interval, the total nodal power uncertainty remains less than the uncertainty assumed in the thermal analysis basis safety limit, maintaining the accuracy of the thermal limit calculation. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency, and the consequences of an accident previously evaluated are unchanged.

These changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. Therefore, the proposed changes do 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 physically alter the plant or its mode of operation. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing are consistent with current safety analysis assumptions. Therefore, the proposed changes will 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.

There is no impact on equipment design or fundamental operation, and there are no changes being made to safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. The margin of safety can be affected by the thermal limits existing prior to an accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. The thermal limit calculation is not significantly affected because LPRM sensitivity with exposure is well defined. LPRM accuracy remains within the total nodal power uncertainty assumed in the thermal analysis basis, thus maintaining thermal limits and the safety margin.

Since the proposed changes do not affect safety analysis assumptions or initial conditions, the margins of safety in the safety analyses are maintained. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: May 4, 2000, as supplemented May 9, 2000.

Description of amendment request: The proposed amendment would remove the individual control building isolation and recirculation damper numbers from Technical Specification 4.12.1.3 and instead specify ''required'' dampers. The requirement to test these dampers remains the same. The Bases have been modified to indicate that the damper numbers for control building isolation and recirculation are contained in the Updated Final Safety Analysis Report.

Date of publication of individual notice in **Federal Register**: May 22, 2000 (65 FR 32132).

Expiration date of individual notice: June 21, 2000.

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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of application for amendment: April 14, 2000, as supplemented April 20, 2000.

Brief description of amendment: The amendment changed Technical Specification Surveillance Requirement 3.1.3.3 to allow partial insertion of control rod 26–47 instead of insertion of one complete notch. This revised acceptance criterion is limited to the current Unit No. 1 operating cycle, after which the original one-notch requirement will be re-established.

Date of issuance: May 23, 2000.

Effective date: May 23, 2000.

Amendment No.: 210.

Facility Operating License No. DPR–71: Amendment changes the Technical Specifications.

Date of initial notice in **Federal Register**: April 21, 2000 (65 FR 21481). The April 20, 2000, supplemental letter contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 2000.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: July 29, 1999, as supplemented by letter dated January 31, 2000.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.4.9 applicability from Mode 3 with steam dome pressure less than residual heat removal cut in permissive to Mode 3 with steam dome pressure less than 48 psig. Notes associated with TS Surveillance Requirements 3.4.9.1 and 3.5.1.2 are changed to reflect the new 48 psig limit.

Date of issuance: May 23, 2000.

Effective date: May 23, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 164.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 25, 1999 (64 FR 46430).

The January 31, 2000, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 2000.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: July 29, 1999, as supplemented by letters dated August 30, 1999, and February 28, 2000.

Brief description of amendment: The amendment deletes item 3.(b) of Attachment 2 to License Condition 2.C.(16), that required installation of a neutron flux monitoring system, in the form of excore wide range monitors, in conformance with Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.''

Date of issuance: May 18, 2000.

Effective date: May 18, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 162.

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

Date of initial notice in **Federal Register:** October 20, 1999 (64 FR 56530).

The February 28, 2000, supplemental letter provided additional clarifying information but did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 2000.

No significant hazards consideration comments received: No.

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

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: November 18, 1999, as supplemented by a letter dated February 7, 2000.

Brief description of amendment: The amendment revised Subsection 4.3.1.2.b of Technical Specification 4.3, ''Fuel Storage.'' The change revised the wording which described the spacing of the fuel in the new fuel racks.

Date of issuance: May 23, 2000.

Effective date: May 23, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 163.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 29, 1999 (64 FR 73088)

The February 7, 2000, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May, 23, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 29, 1999.

Brief description of amendment: The amendment relocated the requirements associated with the high steam generator level trip functions of the Reactor Protection System from the Technical Specifications to the Technical Requirements Manual.

Date of issuance: May 18, 2000.

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

Amendment No.: 216.

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

Date of initial notice in **Federal Register:** February 9, 2000 (65 FR 6404).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 4, 1999, as supplemented by letter dated May 18, 2000.

Brief description of amendment: The proposed change modifies the Technical Specifications (TS) to extend allowed outage time (AOT) to seven days for one inoperable low pressure safety injection (LPSI) train. Additionally, an AOT of 72 hours is imposed for other conditions where the equivalent of 100 percent emergency core cooling system (ECCS) subsystem flow is available. If 100 percent ECCS flow is unavailable due to two inoperable LPSI trains, an ACTION has been added to restore at least one LPSI train to OPERABLE status within one hour or place the plant in HOT STANDBY within six hours, and to exit the MODE of applicability within the following six hours. In the event the equivalent of 100 percent ECCS subsystem flow is not available due to other conditions, TS 3.0.3 is entered. The Limiting Condition for Operation terminology is being changed for consistency with the ECCS requirements. Additionally, the associated TS Bases are being changed.

Date of issuance: May 25, 2000.

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

Amendment No.: 164.

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

Date of initial notice in **Federal Register:** January 26, 2000 (65 FR 4278).

The May 18, 2000, supplement did not expand the scope of the application as noticed or change the proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 9, 1999, as supplemented by submittals dated March 1, March 13, and May 11, 2000.

Brief description of amendment: This amendment increases the present 100 percent authorized rated thermal power level of 3579 megawatts thermal to 3758 megawatts thermal. This represents a power level increase of 5 percent for the Perry Nuclear Power Plant.

Date of issuance: June 1, 2000.

Effective date: As of the date of issuance and shall be implemented within 90 days.

Amendment No.: 112.

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

Date of initial notice in **Federal Register:** November 3, 1999 (64 FR 59802)

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 1, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 18, 1999, as supplemented September 15, 1999, and March 16, 2000.

Brief description of amendment: The amendment revises Duane Arnold Energy Center (DAEC) Technical Specification (TS) Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' by deleting the manual initiation function of the high pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system isolation. A related condition as well as corresponding surveillance requirements and bases are also deleted.

Date of issuance: June 1, 2000.

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

Amendment No.: 231.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 7, 1999 (64 FR 17026).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 1, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The amendment makes several editorial and administrative changes to the following sections of the Technical Specifications (TSs), Index Page vi, ''Figures 3.4–2 and 3.4–3''; Index Page xv, ''6.0 Administrative Controls''; 4.2.4.2b, ''Determination of Quadrant Power Tilt Ratio''; 6.4.1.7b, ''SORC Responsibilities''; 6.4.2.2d, ''Station Qualified Reviewer Program''; 6.3.1, ''Training''; 6.4.3.9c, ''Records of NSARC''; 6.8.1.6.b.1, ''Core Operating Limits Report''; and 6.8.1.6.b.10, ''Core Operating Limits Report''. In addition, the following Bases sections have been revised: Bases 2.2.1, ''Reactor Trip System Instrumentation Setpoints''; Bases 3/4.2.4, ''Quadrant Power Tilt Ratio''; Bases 3/4.2.5, ''DNB Parameters''; Bases 3/4.4.8, ''Specific Activity''; and Bases 3/4.5.1, ''Accumulators''.

Date of issuance: May 22, 2000.

Effective date: As of its date of issuance, and shall be implemented within 90 days.

Amendment No.: 70.

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

Date of initial notice in **Federal Register:** February 10, 1999 (64 FR 6700).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 12, 2000.

Brief description of amendment: The amendment corrects a reference in Technical Specification Section 6.9.1.8b.1, ''Core Operating Limits Report.''

Date of issuance: May 26, 2000.

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

Amendment No.: 246.

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

Date of initial notice in **Federal Register:** April 21, 2000 (65 FR 21486).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 26, 2000.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: May 19, 2000.

Brief description of amendments: The amendments delay implementation of the improved Technical Specifications to June 30, 2000 from May 31, 2000.

Date of issuance: May 24, 2000.

Effective date: May 24, 2000.

Amendment Nos.: Unit 1—141; Unit 2—141.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised Appendix D of the Operating Licenses.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 24, 2000.

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of application for amendments: May 26, 1999.

Brief description of amendments: The amendments remove Technical Specification (TS) Surveillance Requirement 4.1.3.5.b, control rod scram accumulators' alarm instrumentation, and relocate it to the Updated Final Safety Analysis Report and plant procedures; and revise TS Action Statement 3.1.3.5.a.2.a to allow for an alternate method of determining whether a control rod drive pump is operating.

Date of issuance: May 22, 2000.

Effective date: The amendments are effective as of the date of their issuance and shall be implemented within 30 days. In addition, the licensee shall include the relocated information in the Updated Final Safety Analysis Report submitted to the NRC, pursuant to 10 CFR 50.71(e), as was described in the licensee's application dated May 26, 1999 and evaluated in the staff's safety evaluation dated May 22, 2000.

Amendment Nos.: 143 and 105.

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

Date of initial notice in **Federal Register:** March 22, 2000 (65 FR 15382).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 22, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 27, 1998, as supplemented by letter dated July 1, 1999.

Brief description of amendment: The amendment revises the Permanently Defueled Technical Specifications to delete the requirement for defueled emergency plan procedures. This amendment is contingent upon the transfer of the nuclear spent fuel from the existing 10 CFR Part 50 licensed area to the 10 CFR Part 72 independent spent fuel storage installation area.

Date of issuance: May 10, 2000.

Effective date: May 10, 2000, and shall be implemented within 30 days after the transfer of the last cask of spent nuclear fuel from the spent fuel pool to the independent spent fuel storage installation is complete.

Amendment No.: 202.

Facility Operating License No. NPF–1: The amendment changes the Permanently Defueled Technical Specifications.

Date of initial notice in **Federal Register:** August 25, 1999 (64 FR 46441).

The July 1, 1999, supplemental letter provided additional clarifying information and did not expand the scope of the application as originally noticed and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 2000.

No significant hazards consideration comments received: No.

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: March 14, 2000, as supplemented March 27, and May 25, 2000.

Brief description of amendments: The amendments extended the implementation date for Amendment No. 184 to Facility Operating License NPF–14 and Amendment No. 158 to Facility Operating License NPF–22 from 30 days following startup from the Unit 1 Spring 2000 refueling outage to no later than November 1, 2001.

Date of issuance: June 2, 2000.

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

Amendment Nos.: 187 and 161.

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

Date of initial notice in **Federal Register:** April 27, 2000 (65 FR 24718). The May 25, 2000, letter provided clarifying information but 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 June 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 3, 2000, as supplemented on May 19, 2000.

Brief description of amendment: The license amendment modifies the existing requirement under Technical

Specification Section 3.1.3.2.1, Action a.1, to determine the position of Rod 1SB2 from once every 8 hours to within 8 hours following any movement of the rod until repair of the rod indication system is completed. This change is applicable for the remainder of the Unit 1 Cycle 14, or until an outage of sufficient duration occurs whereby the licensee can repair the position indication system.

Date of issuance: May 26, 2000.

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

Amendment No.: 230

Facility Operating License No. DPR–70: This amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (65 FR 30137) May 10, 2000. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. That notice also provided for an opportunity to request a hearing by May 24, 2000, 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 is contained in a Safety Evaluation dated May 26, 2000.

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

Date of application for amendment: January 27, 2000.

Brief description of amendment: The amendment revises the spent fuel pool reactivity limit requirement by removing the value for K infinity from Specification 5.6.1.1 and replacing it with a figure of integral fuel burnable absorbers rods versus nominal Uranium-235 enrichment.

Date of issuance: June 1, 2000.

Effective date: June 1, 2000.

Amendment No.: 144.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** February 23, 2000 (65 FR 9011).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 1, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 15, 2000.

Description of amendment request: The amendments revised the Technical Specifications (TS) to provide a 7-day limiting condition for operation when two trains of the Containment Air Dilution System are inoperable.

Date of issuance: May 24, 2000.

Effective date: May 24, 2000.

Amendment Nos.: 265 and 225.

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

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17919).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 24, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 29, 1999.

Brief description of amendments: These amendments revise Technical Specification (TS) Section 3/4.3.3, ''Radiation Monitoring Instrumentation,'' TS Section 3/4.7.7, ''Control Room Emergency Ventilation System,'' and the associated bases. Actions are added and modified regarding inoperable equipment.

Date of issuance: May 31, 2000.

Effective date: May 31, 2000.

Amendment Nos.: 256 and 247.

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

Date of initial notice in **Federal Register:** May 19, 1999 (64 FR 27325).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 6, 2000 (ULNRC–04197).

Brief description of amendment: The amendment revises Limiting Condition for Operation (LCO) 3.7.1, ''Main Steam Safety Valves (MSSVs),'' in that the maximum allowable reactor power for a given number of operable MSSVs per steam generator is reduced in Table 3.7.1–1, ''Operable Main Steam Safety Valves [MSSVs] versus Maximum Allowable Power,'' and in Required Action A.1 of the TSs. These changes will result in decreasing the setpoint values for the power range neutron flux high channels, which are part of the reactor trip system (RTS) instrumentation in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' and will result in the reactor operating at a lower power for a given number of operable MSSVs per steam generator. In addition, two format errors in the actions for LCO 3.7.1 are corrected.

Date of issuance: May 26, 2000.

Effective date: May 26, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 136.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17920).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 26, 2000.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By July 14, 2000, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of amendment request: May 12, 2000, as supplemented by letter dated May 19, 2000.

Brief description of amendment: The amendment revises TS 3.7.3, Condition A, to extend the Completion Time for one or more feedwater isolation valves (FIVs) inoperable from 4 hours to 24 hours if, within 4 hours, the respective feedwater control valves (FCVs) and the FCV bypass valves in the same flowpath are verified to be capable of performing the feedwater isolation function. A footnote is added that indicates that the extension of the Completion Time to 24 hours is only applicable for repair of the FIV hydraulic system through fuel cycle 8 for Unit 1 and fuel cycle 5 for Unit 2.

Date of issuance: May 25, 2000.

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

Amendment No.: 77.

Facility Operating License Nos. NPF–87 and NPF–89: The amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on May 24, 2000. The notice was published in the Dallas Morning News and the Ft. Worth Star Telegram from May 21 through May 23, 2000.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Texas, and final no significant hazards consideration determination are contained in a Safety Evaluation dated May 25, 2000.

Dated at Rockville, Maryland, this 7th day of June 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–14837 Filed 6–13–00; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

State of Oklahoma: NRC Staff Assessment of a Proposed Agreement Between the Nuclear Regulatory Commission and the State of Oklahoma

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of a proposed Agreement with the State of Oklahoma.

SUMMARY: This notice is announcing that the Nuclear Regulatory Commission (NRC) has received a request from Governor Frank Keating of Oklahoma that the NRC consider entering into an Agreement with the State as authorized by Section 274 of the Atomic Energy Act of 1954, as amended (Act). Section 274 of the Act contains provisions for the Commission to enter into agreements with the Governor of any State providing for the discontinuance of the regulatory authority of the Commission. Under the proposed Agreement, submitted December 28, 1999, the Commission would discontinue and Oklahoma would take over portions of the Commission's regulatory authority over radioactive material covered under the Act within the State of Oklahoma. In accordance with 10 CFR 150.10, persons, who possess or use certain radioactive materials in Oklahoma, would be released (exempted) from portions of the Commission's regulatory authority under the proposed Agreement. The Act requires that NRC publish those exemptions. Notice is hereby given that the pertinent exemptions have been previously published in the **Federal Register** and are codified in the Commission's regulations as 10 CFR Part 150. NRC is publishing the proposed Agreement for public comment, as required by the Act. NRC is also publishing the summary of an assessment conducted by the NRC staff of the proposed Oklahoma byproduct material regulatory program. Comments are invited on (a) the proposed Agreement, especially its effect on public health and safety, and (b) the NRC staff assessment.

DATES: The comment period expires July 7, 2000. Comments received after this date will be considered if it is practical to do so, but the Commission cannot assure consideration of comments received after the expiration date.

ADDRESSES: Written comments may be submitted to Mr. David L. Meyer, Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Washington, DC 20555–0001. Copies of comments received by NRC may be examined at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Copies of the proposed Agreement, copies of the request for an Agreement by the Governor of Oklahoma including all information and documentation submitted in support of the request, and copies of the full text of the NRC staff assessment are also available for public inspection in the NRC's Public Document Room.

FOR FURTHER INFORMATION CONTACT: Patricia M. Larkins, Office of State and Tribal Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Telephone (301) 415–2309 or e-mail *pml@nrc.gov*.

SUPPLEMENTARY INFORMATION: Since Section 274 of the Act was added in 1959, the Commission has entered into Agreements with 31 States. The Agreement States currently regulate approximately 16,000 agreement material licenses, while NRC regulates approximately 5800 licenses. Under the proposed Agreement, approximately 220 NRC licenses will transfer to Oklahoma. NRC periodically reviews the performance of the Agreement States to assure compliance with the provisions of Section 274. Section 274e requires that the terms of the proposed Agreement be published in the **Federal Register** for public comment once each week for four consecutive weeks. This notice is being published in fulfillment of the requirement.

I. Background

(a) Section 274d of the Act provides the mechanism for a State to assume regulatory authority, from the NRC, over certain radioactive materials [1] and activities that involve use of the materials. In a letter dated December 28,

[1] The radioactive materials, sometimes referred to as agreement materials, are: (a) Byproduct materials as defined in Section 11e.(1) of the Act; (b) byproduct materials as defined in Section 11e.(2) of the Act; (c) source materials as defined in Section 11z. of the Act; and (d) special nuclear materials as defined in Section 11a. of the Act, restricted to quantities not sufficient to form a critical mass.