for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants shall be prohibited.

(c) No disbursements or payments from the trust shall be made by the trustee until the trustee has first given the NRC 30 days notice of the payment. In addition, no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director, Office of Nuclear Reactor Regulation.

(d) The trust agreement shall not be modified in any material respect without prior written notification to the Director, Office of Nuclear Reactor Regulation.

(e) The trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a ''prudent investor'' standard, as specified in 18 CFR 35.32(3) of the Federal Energy Regulatory Commission's regulations.

6. PSEG Nuclear shall not take any action that would cause PSEG Power LLC or its parent companies to void, cancel, or diminish the commitment to fund an extended plant shutdown as represented in the application for approval of the transfer of the HCGS license from PSE&G to PSEG Nuclear.

7. Before the completion of the transfer of the interest in HCGS to PSEG Nuclear as previously described herein, PSEG Nuclear shall provide to the Director, Office of Nuclear Reactor Regulation, satisfactory documentary evidence that PSEG Nuclear has obtained the appropriate amount of insurance required of licensees under 10 CFR Part 140 of the Commission's regulations.

8. After receipt of all required regulatory approvals of the subject transfer, PSE&G shall inform the Director, Office of Nuclear Reactor Regulation, in writing of such receipt, and of the date of closing of the transfer no later than seven business days prior to the date of closing. Should the transfer not be completed by December 31, 2000, this Order shall become null and void, provided, however, on application and for good cause shown, such date may be extended.

It is further ordered that, consistent with 10 CFR 2.1315(b), a license amendment that makes changes, as indicated in Enclosure 2 to the cover letter forwarding this Order, to conform the license to reflect the subject license transfer is approved. Such amendment shall be issued and made effective at the time the proposed license transfer is completed.

This Order is effective upon issuance.

For further details with respect to this Order, see the initial application dated June 4, 1999, and the supplement dated October 22, 1999, which are available for public 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 16th day of February 2000.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 00–4256 Filed 2–22–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 29, 2000, through February 11, 2000. The last biweekly notice was published on February 9, 2000 (65 FR 6402).

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 March 24, 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).

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

Date of amendment request: May 27, 1999.

Description of amendment request: The requested amendment proposes to increase the maximum allowable Service Water (SW) temperature used to determine operability of the Ultimate Heat Sink (UHS) from 95 °F to 97 °F. The amendment includes all the TS changes necessary as a result of new analyses performed to support the increase of the maximum allowable SW temperature.

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:

Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change increases the maximum allowable Service Water (SW) temperature, which is used to determine OPERABILITY of the Ultimate Heat Sink (UHS), from 95 °F to 97 °F. As a result of the new analyses to support the increase in SW temperature, the proposed change also decreases the required actuation setpoint for the Containment Pressure High High signal from 20 psig to 10 psig, decreases the closure time credited for the Main Feedwater

Isolation Valves (MFIVs) in the analysis from 80 seconds to 50 seconds, increases the required operating pressure for the Isolation Valve Seal Water (IVSW) and IVSW nitrogen bottle pressure from 44 psig to 44.6 psig, decreases the closure time for Main Steam Isolation Valves (MSIVs) credited in the analysis from 5 seconds to 2 seconds, and increases the peak calculated containment internal pressure for a large break Loss of Coolant Accident (LOCA), Pa, from 40 psig to 40.5 psig. In addition, the Containment Spray (CS) actuation circuitry will be modified to allow the CS pumps to be restarted after they have been stopped while the original actuation signal is present.

SW temperature is not itself an initiator of accidents evaluated in the Safety Analysis report (SAR). The components provided SW flow that are required to perform a safety-related function are designed to operate at temperatures above the temperatures to which SW will be increased. Therefore, these components are not more likely to fail and initiate an accident. The components have been shown to perform their intended safety related function with the higher SW temperatures. Containment analyses have been performed that show that containment integrity and equipment environmental qualification are maintained.

The modification to the Containment High High Pressure actuation setpoint will not increase the probability of an unwanted actuation. Changing the actuation setpoint will not change the reliability of this function. The Containment Pressure High High Pressure function will (1) initiate Containment Spray sooner, which will mitigate the pressure and temperature transient sooner, and (2) isolate leakage of radioactivity from containment through ''essential'' process lines sooner in an accident. Also, the lower actuation setpoint, in conjunction with other analysis assumptions, has been evaluated to result in a slight decrease (−2 °F) in the large break LOCA Peak Cladding Temperature.

Crediting faster MFIV closure in the Main Steam Line Break (MSLB) containment analysis will not change the probability of MFIV failure or the probability that the MFIV will initiate an accident because a physical modification is not associated with the proposed change. (The physical modification is being implemented in accordance with 10 CFR 50.59). Since there is no physical modification, the amount of feedwater addition to containment during [an] MSLB if the Main Feedwater Regulating Valve (MFRV) fails [to] open will not change, although the amount calculated by the analysis will be reduced.

Crediting faster MSIV closure in the MSLB containment analysis will not change the probability of MSIV failure or the probability that the MSIV will initiate an accident because a physical modification is not involved. Since there is no physical modification, the amount of blowdown from the unaffected SGs [steam generators] and the amount of radioactivity released to the environment by [an] MSLB will not be adversely affected, although the amount calculated by the analysis will be reduced. Crediting a faster closure time does not require crediting a faster MSIV opening time because of the valve design, and opening [an] MSIV is not postulated for an analyzed accident.

Changing the minimum operating pressure of the IVSW components does not involve a physical modification, hence, will not affect the probability that components will fail or initiate an accident. The IVSW system will perform its containment isolation function by providing a water seal at the higher pressure calculated by the new large break LOCA containment analysis.

The Containment Leakage Rate Testing (CLRT) program historically has performed integrated leak rate testing and local leak rate testing at pressures higher than the peak containment pressure calculated by the new large break LOCA containment analysis. The components which are tested by the CLRT program are designed for operation at a pressure higher than the pressure to which they are tested. The current CLRT program ensures that the containment leakage is less than that used to calculate the doses for a large break LOCA accident.

The modification to the CS actuation circuitry will not affect the reliability of the circuit. The modification will be tested periodically to ensure reliability and to confirm the capability of restoring CS after being blocked. Blocking the actuation circuitry will be procedurally controlled and will allow the CS pumps to be restarted, after being stopped, when an actuation signal is present. The analysis results show that containment pressure and temperature are within design limits when CS is stopped for the switchover.

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

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

The components provided SW flow have been shown to perform their safety related function with the higher service temperature, hence, will not exhibit any new type of failure mechanism or mode as a result of the increased temperatures.

Decreasing the Containment High High Pressure actuation setpoint only changes the time at which the signal is generated, not how it is generated, or how the actuated equipment responds to the signal, hence, will not introduce any new types of failures.

Crediting faster MFIV and MSIV stroke times in the MSLB containment analysis does not involve a physical modification, hence, can not introduce any new failure modes.

The IVSW components and the components tested by the CLRT program are designed for pressures that are higher than the pressures at which they are proposed to operate and be tested. As the functions of these components are not changing, and the components are capable of withstanding the higher pressure, a higher operating or testing pressure will not create any new failure mechanisms or accidents.

The modification to the CS actuation circuitry will be tested periodically to ensure proper operation and reliability of the circuit. Even if one of the blocking circuits should fail during operation, a single failure of a CS pump has been considered in the containment analysis, hence, is not a new type of failure or accident.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

Containment structural integrity, containment leakage, fuel cladding, equipment environmental qualification, EDG electrical capacity, and UHS capability were considered to determine if the proposed change involves a significant reduction in a margin of safety.

Containment pressure is limited to the design pressure of 42 psig to maintain structural integrity. A structural integrity test at 115% of the design pressure (48.3 psig) has confirmed the containment's structural capability. The new containment analyses for large break LOCA and MSLB using [an] SW temperature of 100 °F show that the containment pressure does not exceed 42 psig. The margin of safety for containment is not reduced by the proposed change because the design pressure is not exceeded. The containment leakage rate, La, is limited to 0.1% of the containment air weight per day. La is based on the peak calculated containment internal pressure, Pa, for the design basis LOCA. The offsite doses resulting from an accident are based on La. If containment leakage does not exceed La, the margin of safety is not reduced. The leakage rates for Type A, B, and C containment penetrations are measured periodically throughout plant life to ensure that containment leakage is [less than or equal to] La. The leakage rate acceptance criteria are [less than or equal to] 0.75 L for Type A tests, and [less than or equal to] 0.60 La for Type B and Type C tests. As a result of using [an] SW temperature of 97 °F in the new large break LOCA containment analysis, Pa has changed from 40 psig to 40.5 which changes the pressure at which the Type A, B, and C containment penetration leakage is measured. Historically, containment leakage rate testing has been performed at the containment design pressure of 42 psig or higher. The margin of safety related to containment leakage is not reduced by the proposed change because containment leakage is [less than or equal to] La.

Fuel cladding integrity is evaluated by determining the effect on the Peak Cladding Temperature (PCT) and the Departure to Nucleate Boiling Ratio (DNBR) for postulated accident. The PCT for a large break LOCA changes by −2 °F as a result of the proposed change including associated changes. The DNBR for a non-limiting case of the MSLB changes, but the margin to the DNBR limit is very large. Therefore, fuel cladding integrity is not adversely affected.

Safety-related equipment is potentially required to function in an adverse environment during and following an accident. Using [an] SW temperature of 97 °F, the new large break LOCA and MSLB containment analyses yield temperature and pressure profiles show that the temperature and pressure profiles for equipment required to operate during and following an accident are qualified. The margin of safety related to equipment environmental qualification is not reduced by the proposed change because equipment required to operate during and following an accident are environmentally qualified.

The Emergency Diesel Generators (EDGs) provide emergency electrical power to run safety-related equipment following an accident that is accompanied by a loss of offsite power. The EDGs are rated at 110% capacity for 2 hours out of each 24 hours and tested between 106% to 110% for at least 1.75 hours. Since the EDG can provide 110% for 1.75 hours, the margin of safety is not reduced. Using [an] SW temperature of 97 °F, a calculation shows that adequate cooling is provided for the EDG to produce 110% electrical output.

The UHS is required to provide cooling water for at least 22 days following a design basis accident. The UHS is able to provide cooling water for 22.1 days at a temperature of 100 °F. Therefore, the cooling capability of the UHS would not be adversely affected.

Based on the above, it may be concluded that 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 22, 1999.

Description of amendment request: The proposed amendment would expand the Core Operating Limits Report (COLR) and relocate reactor coolant system related cycle-specific parameter limits from the technical specifications (TSs) and include them in the COLR.

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

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

The proposed changes are programmatic and administrative in nature which do not physically alter safety-related systems, nor affect the way in which safety-related systems perform their functions. The proposed changes remove cycle-specific parameter limits from TS 3.4.1 and relocate them to the COLRs which do not change plant design or affect system operating parameters. In addition, the minimum limit for [Reactor Coolant System] RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. The existing TS Section 5.6.5b, COLR Reporting Requirements, continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. The existing TS Section 5.6.5c, COLR Reporting Requirements, continues to ensure that applicable limits of the safety analyses are met. Further, more specific requirements regarding the safety limits (*i.e.,* [Departure from Nucleate Boiling Ratio] DNBR limit and peak fuel centerline temperature limit) are being imposed in TS 2.1.1, ''Reactor Core Safety Limits,'' replacing the Reactor Core Safety Limits (RCSL) figure which are consistent with the values stated in the Updated Final Safety Analysis Report (UFSAR).

Although the relocation of the cycle-specific parameter limits to the COLRs would allow revision of the affected parameter limits without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits. The differences would not be significant and would be bounded by the existing requirement of TS Section 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameter limits being transferred from the TS to the COLRs will continue to be controlled under existing programs and procedures. The UFSAR accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

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

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

The proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameter limits from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, other than their relocation to the COLRs. There are no setpoints affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

Relocation of cycle-specific parameter limits has no influence or impact on, nor does it contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameter limits will continue to be calculated using the NRC reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis and operation within the core operating limits will continue.

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor does it effect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameter limits to the COLRs will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to operate within prescribed limits.

The development of cycle-specific parameter limits for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that plant operation within cycle-specific parameter limits will not involve a significant reduction in the margin of safety.

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

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

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

Date of amendment request: September 17, 1999.

Description of amendment request: The proposed amendment would change the Arkansas Nuclear One, Unit 2 (ANO–2) heavy load handling requirements and transportation provisions to permit the movement of the original and replacement steam generators through the ANO–2 containment construction opening during the steam generator replacement 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:

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

During the 2R14 refueling outage/steam generator replacement outage, the OSGs [original steam generators] and the RSGs [replacement steam generators] will be moved between the new steam generator storage area/original steam generator storage facility and the runway beam support system (RBSS)/outside lift system (OLS). The RBSS/OLS is the structure used to rig the SGs [steam generators] in and out of the reactor containment building. In consideration of the magnitude of the loads being handled, the RBSS, OLS and transporters are of a robust, rugged design, proven by many prior steam generator replacements and other heavy load handling operations. However, due to the location of safety related underground structures, systems, and components (SCCs) in the vicinity of the RBSS/OLS and along the steam generator (SG) haul route, potential load handling accidents along the load paths must be considered for their effects on the SCCs. At ANO–2, the ground cover over several buried SSCs is not sufficient to be able to rule out the potential for a load drop to damage or cause failure of these SCCs. The functions of the SSCs in question are as support systems to the ANO–1 [Arkansas Nuclear One, Unit 1] and ANO–2 emergency diesel generators and the ANO–1 service water system. The fire protection system, a non-safety related system, was also considered. Existing plant procedures adequately address the scenario in question for the fire protection system.

The cause of a SG drop is assumed to be a non-mechanistic failure of the RBSS/OLS (or associated rigging), a failure of the SG transporter leveling hydraulics, or a seismically-induced failure of the loaded RBSS/OLS or SG transporter. The possibility of drops associated with other external events, such as tornadoes, high winds, and tornado missiles will be substantially minimized by procedures that prevent load handling under these weather conditions.

With ANO–2 defueled, the impact on ANO–2 due to loss of the emergency diesel generators fuel oil transfer system will be minimal. Long term actions to provide makeup water to the spent fuel pool may be necessary, but no immediate actions are required.

For ANO–1, a steam generator drop could render both diesel generators inoperable due to the loss of the fuel oil transfer system, and the emergency cooling pond inoperable due to the loss of the service water return line to the pond. Since ANO–1 is expected to be at full power operation, these conditions would require prompt action in accordance with technical specifications. Immediately following a drop from the OLS or from the transporter in the vicinity of the OLS, where damage to these systems is possible, ANO–1 will begin a shutdown and cooldown to cold shutdown conditions. In conjunction with the unit shutdown, contingency actions to provide temporary connections from the fuel oil storage facility to the ANO–1 emergency diesel generator day tanks, and temporary power to the fuel transfer pumps would be implemented.

The ability of ANO–1 to safely respond to analyzed events would be undiminished with the possible exception of the functions affected by the damaged equipment. With the compensatory measures to be established prior to the steam generator handling operations, and with the planned responses to a steam generator drop, the support system functions of the diesel generators and the service water system can be assumed to be maintained following the drop. Therefore, the drop will not affect the consequences of any analyzed event.

While the drop of a steam generator could cause damage to some safety related plant equipment, the failures of these components are not precursors to any analyzed accident. The drop of a steam generator will not have any other impact on plant equipment, and thus will not induce any analyzed plant transient. It will, however, result in a malfunction of equipment important to safety of a different type than any previously evaluated. Based on the compensatory measures and the low likelihood of the event during SG movement, this temporary condition is considered to be acceptable. On these bases, it is concluded that the proposed load handling operations will not significantly increase the probability or the consequences of accidents previously analyzed.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

As noted in the response to the first question above, the only potential for a new or different kind of accident associated with this change request arises from a drop of a steam generator which is assumed to cause the loss of emergency power support systems for ANO–1. The cause of a SG drop is assumed to be a non-mechanistic failure of the RBSS/OLS (or associated rigging), a failure of the SG transporter leveling hydraulics, or a seismically-induced failure of the loaded RBSS/OLS or SG transporter. In the absence of a seismic event, there is no initiator for any consequential events (*e.g.,* loss of offsite power) other than those directly caused by impact of the SG. Given this scenario, the plant response to a SG drop event would be governed by the technical specifications and existing plant procedures.

If a SG drop is seismically-induced, the simultaneous loss of normal offsite power sources is also assumed in this case since these sources are not seismically qualified. While this event is very unlikely due to the low frequency of earthquakes and the small amount of time that a steam generator will be in a position to cause damage, Entergy [Operations, Inc.] will provide contingency plans and compensatory measures so that makeup to the ANO–2 spent fuel pool and fuel oil supply to the ANO–1 emergency diesel generators and transfer pump power supply are assured under any circumstances.

Availability of the redundant ANO–1 service water heat sink, the Dardanelle Reservoir, during a seismic event assures that an uninterrupted source of service water will be available to support shutdown cooling of ANO–1.

The proposed load handling plans will not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety

ANO–1 Technical Specification 3.7.1.C requires both EDGs [emergency diesel generators] to be operable when the reactor temperature is ≥200 °F. If this condition is not met, Limiting Condition for Operation 3.0.3 applies. It requires that within one hour, action shall be initiated to place the unit in an operating condition in which the specification does not apply by placing it, as applicable, in at least hot standby within the next 6 hours, at least hot shutdown within the following 6 hours, and at least cold shutdown within the subsequent 24 hours. The bases for technical specification 3.7.1.C indicate that these operability requirements ensure that an adequate, reliable power source is available for all electrical equipment during startup, normal operation, safe shutdown, and handling of all emergency situations. The bases for EDG operation also require at least a seven day total diesel oil inventory during complete loss of electrical power conditions.

The postulated loss of both trains of the ANO–1 EDG fuel oil transfer system due to a SG drop would require that ANO–1 be shut down. This situation could be considered to involve a reduction in the margin of safety, because a new common cause failure mechanism is being introduced by the movement of the SGs over the EDG fuel oil lines and transfer pump power cables. To restore the margin of safety and return the EDGs to functionality, temporary compensatory measures are being proposed.

Based on the above discussions, with the implementation of the proposed compensatory measures and the low likelihood of such an event, the failures caused by a SG drop event 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: Nicholas S. Reynolds, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 18, 1999.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 4.4.5, ''Steam Generators,'' to note that the requirements for inservice inspection do not apply during the steam generator replacement outage (2R14), to delete inspection requirements associated with steam generator tube sleeving and repair limits, to extend the inspection interval to a maximum of once per 40 months provided the inspection results from the first inspection following the preservice inspection fall into the C–1 category, to revise the preservice inspection requirements on when the hydrostatic test and the eddy current inspection of the tubes would be performed, and to revise the reporting frequency of the results of steam generator tube inspections to within 12 months following completion of the inservice inspection. Related changes to the Bases would also be made.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The accidents of interest are a tube rupture, loss of coolant accident (LOCA) in combination with a safe shutdown earthquake and a steam line break in combination with a safe shutdown earthquake. A reduction in tube integrity could increase the possibility of a tube rupture accident and increase the consequences of a steam line break or LOCA. The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code], Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action if tube degradation is present. The ASME [Code], Section XI basis for the 40% through wall plugging limit is applicable to the replacement steam generators just as it was to the original steam generators. As a result there is no reduction in tube integrity for the replacement steam generators.

Addition of a ''Note'' to clarify that inservice inspection is not required during the steam generator replacement outage is an administrative change that provides clarification regarding inservice inspection requirements. The change in reporting requirements is also an administrative change. The requirements for inservice inspection or the plugging limit for the tubes are not altered by these administrative changes. Additionally, changes were made to the bases to remove potentially misleading information. Bases changes are considered to be administrative in nature.

Elimination of the repair option and the associated references to repair of the original steam generator tubes is an administrative adjustment since the sleeve design is not applicable to the replacement steam generators. The elimination of the repair option does not alter the requirements for inservice inspection or reduce the plugging limit for the tubes.

The proposed change to extend the inspection interval to a maximum of once per 40 months is acceptable based on the use of the superior Alloy 690 tubing material. Significant industry knowledge has been gained from monitoring the performance of steam generators that have been replaced. Alloy 690 tubing material has proven to be superior to Alloy 600 in regard to corrosion resistance. Plants that have utilized Alloy 690 tubing in their replacement steam generators have not experienced corrosion-induced degradation.

A preservice eddy current inspection will be performed onsite prior to installation of the replacement steam generators. The orientation of the replacement steam generators during the eddy current exam will not impact the results. The hydrostatic test required by the ASME Code, Section III for the replacement steam generators is to be performed in the manufacturing facility and not as part of a reactor coolant system hydrostatic test. The post-repair leakage test required by the ASME Code, Section XI for an operating plant is performed at a much lower pressure. No evolutions subsequent to the replacement steam generator hydrostatic test are expected to occur that will change the condition of the tubes prior to operation. This change does not alter the requirement to perform a preservice inspection. As a result, an inservice inspection is not required during the steam generator replacement outage.

The requested ANO–2 [Arkansas Nuclear One, Unit 2] Technical Specification changes do not alter the requirements for tube integrity, tube inspection, or tube plugging limit. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed changes do not affect the design or function of any other safety-related component. There is no mechanism to create a new or different kind of accident for the replacement steam generators by eliminating repair criteria or by clarifying the applicable preservice and inservice inspection requirements because a baseline of tube conditions is established and plugging limits are maintained to ensure that defective tubes are removed from service. A change in inspection frequency has a negligible impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building. Changing the inspection frequency creates no new failure modes or accident initiators/precursors.

The requested ANO–2 Technical Specification changes do not alter the requirements for tube integrity, tube inspection or tube plugging limit. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety

The tubing in the replacement steam generators is designed and evaluated consistent with the margins of safety specified in the ASME Code, Section III. The program for periodic inservice inspection provides sufficient time to take proper and timely corrective action to preserve the design margin if tube degradation is present.

Due to the superior Alloy 690 tubing material and the significant amount of industry knowledge and operating history with this improved tubing material, extending the inspection interval to a maximum of once per 40 months will still allow the integrity of the steam generator tubing to be ensured. The steam generator inspection program is not intended to provide an accident mitigation or assessment function; therefore, this change results in a neutral impact to the margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 3, 1999.

Description of amendment request: The proposed amendment would increase the containment structural design pressure from 54 to 59 psig, revise Technical Specification (TS) Table 3.3–3 to add a containment spray actuation signal on high-high containment building pressure to terminate main feedwater and main steam flow from the unaffected steam generator, revise TS 3.6.1.4 and Figure 3.6–1 to change the allowable containment initial conditions to be consistent with analysis assumptions, revise TS 4.6.2.1 to increase the allowable containment spray pump degradation from 6.3% to 10.0%, and revise TS 6.15 to increase the calculated peak accident pressure in the containment leakage rate testing program from 54 to 58 psig and to clarify the allowable leakage rate. Related changes to the Bases would also be made.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The containment building will meet structural requirements for the higher design pressure. Except for the application of CSAS [Containment Spray Actuation Signal] in a different manner than used previously, the electrical penetration seal modifications and the containment cooling fan pitch change, increasing the containment structural design pressure is analytical. There are no changes to the allowable containment leakage rate. The increase in design pressure requires changes to the bases of the technical specifications and the SAR [Safety Analysis Report]. However, the peak accident and design pressures are below the failure pressure of any potentially affected system, structure or component. The change does not increase the probability of an accident previously evaluated. Since the containment leakage rate will not increase, the consequences of any previously evaluated accident will not increase. Therefore, the increase in design and peak pressures does not involve a significant increase in the probability or consequences of an accident previously evaluated.

A structural integrity test (SIT) will be performed at 1.15 times the new design pressure of 59 psig. The SIT will provide acceptance criteria to assure that measured responses are within the limits predicted by analyses.

Additionally, evaluations of components within the containment building demonstrate that the components are qualified to the increased pressure.

Revising the allowable containment operating conditions provides more operating flexibility than current requirements. The proposed change is consistent with the assumptions made in the revised containment peak pressure analyses. Since the change only affects containment atmosphere conditions allowed during normal operation, it has no impact on the probability of initiation of a previously evaluated accident. Therefore, this aspect of the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The increase in peak accident pressure will also require leakage rate testing of the containment structure and its penetrations to be performed at a 4 psi higher pressure than was required previously. Increasing the value of P_a in the containment leakage rate program changes the conditions for performing the tests. Since the revised value is well within the design capabilities of SSCs [systems, structures and components] that could be affected during the performance of the test, it will not weaken any of the protective barriers. Many past local leak rate tests have been performed at increased pressures (59–60 psig) with no significant difference in leakage results. Based on the leakage testing history, no problems are expected from the increase in P_a. Further, since these tests are not performed when the plant is operating, they have no impact on normal plant operation or the outcome of any previously evaluated accident. Therefore, this aspect of the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Revising the allowable degradation of the containment spray pump does not create the probability or consequences of an accident previously evaluated. Although the allowable pump degradation increased from 6.3% to 10%, analysis has shown that at 10% degraded, the pumps can deliver to containment the flow required at 59 psig and required to reduce containment pressure to an acceptably low level.

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

Increasing the containment structural design pressure due [to] replacing the steam generators and the future 7–12% power uprate does not result in the failure of any system, structure or component during the progression of any previously evaluated accident. Therefore, the progression of the previously evaluated accidents will not change. Further, the change in design pressure is primarily administrative and does not affect the way the plant is operated. Therefore, this aspect of the change does not create the possibility of a new or different kind of accident from any previously evaluated.

The added CSAS actuating signal results in isolating the steam generators for events that generate a containment pressure high-high signal. CSAS, a four channel safety grade system, is part of the reactor protection system (RPS). The RPS is designed to reliably mitigate the effects of an accident. The only new condition created by this change would be the isolation of the steam generators upon an inadvertent actuation of CSAS. The possibility of steam generator isolation currently exists for an inadvertent MSIS [Main Steam Isolation Signal]. This condition is not considered to be an accident given the safety grade equipment available to mitigate this event and minor consequences due to its occurrence. The CSAS change will be implemented such that no new or failure modes or effects will be created that could cause a new or different kind of accident from any previously evaluated.

Revising the allowable containment operating conditions permits the plant to be operated for a wider range of containment atmospheric conditions. This aspect of the proposed change reduces the likelihood of a plant upset as a result of shutting the plant down in response to exceeding a limiting condition for operation. The proposed change is consistent with the assumptions made in the accident analysis and will insure that the containment peak pressure and temperature do not exceed design limits following design basis accidents. Therefore, this aspect of the change does not create the possibility of a new or different kind of accident from any previously evaluated.

Revising the value of P_a in the containment leakage rate program changes the conditions for performing the 10 CFR 50 Appendix J leak rate test. The revised value is well within the design capabilities of SSCs that could be affected during the performance of the test. Therefore, this aspect of the change does not create the possibility of a new or different kind of accident from any previously evaluated.

Revising the allowable degradation of the containment spray pump does not increase the possibility of a new or different kind of accident from any previously evaluated. Although the allowable pump degradation increased from 6.3% to 10%, analysis has shown that when degraded 10%, the pumps can deliver the required flow to the containment building at the increased containment pressure of 59 psig.

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety

Increasing the containment structural design pressure from 54 to 59 psig causes a small reduction in the design margin for the containment response. Based on the analyses performed, the reduction has been determined to be acceptable since code allowable stresses are not exceeded. The analyses demonstrate that the containment meets all applicable codes and standards at 59 psig. Since the physical containment structure is not changed as a result of this reanalysis, the stresses on the containment structure following a design basis event are increased as a result of this change. Since the margin of safety is the difference between the stresses that would result in containment failure and the stresses at design conditions, this change involves a reduction in the margin of safety. However, the containment failure pressure is much higher than the design basis accident pressure. Also, the DBA [Design Basis Accident] peak pressure is currently very close to the design pressure. With the proposed change, there is margin between the DBA and design pressures. Therefore, this change does not significantly increase the probability of containment failure for design basis events. The ANO–2 [Arkansas Nuclear One, Unit 2] containment building was designed and constructed using significant conservatisms.

The new application of the CSAS signal is proposed to reduce the severity (*i.e.,* reduce the mass and energy addition) of the increased effect of a main steam line break inside containment. Since this aspect of the proposed change improves the response of the plant to this design basis event, it does not involve a significant reduction in margin of safety.

Revising the allowable containment operating conditions provides additional operating margin. The proposed allowable operating conditions are consistent with the accident analyses performed to demonstrate that the peak containment pressure is less than design pressure. The relaxation in containment operating conditions was made possible by the increase in containment design pressure and the addition of the new CSAS actuation to selected components that previously received only an MSIS actuation signal.

Increasing the value of P_a in the containment leakage rate program changes the conditions for performing the tests. [Fifty-nine] psig is well within the design capabilities [of] SSCs that could be affected by the tests. The leakage rate tests will not weaken any of the protective barriers. Past local leak rate tests have been successfully performed at increased pressures (59–60 psig) with no significant difference in leakage results. Therefore, this aspect of the change does not involve a significant reduction in the margin of safety.

As discussed previously, increasing the allowable containment spray pump degradation does not increase the probability or consequences of an accident previously evaluated. Although the allowable pump degradation increased from 6.3% to 10%, analysis has shown that at 10% degraded, the pumps can deliver the required flow to the containment building at the increased containment pressure of 59 psig.

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy [Entergy Operations, Inc.] has determined that the requested change does not involve a significant hazards consideration.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 27, 2000.

Description of amendment request: The proposed amendment would delete the current requirements of Technical Specification (TS) 4.7.9.1.2.d, ''Source installed in the Boronometer,'' associated with the installed boronometer sealed source. The source was recently removed and stored, and the requirements of TS 4.7.9.1.2.d are no longer applicable.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The modification performed on the boronometer removed its sealed source and placed the source in safe storage. The removal of this source from plant systems removes the possibility of contamination or radiological exposure from this source to personnel working on or near the boronometer. Since the source has been placed in safe storage, no change in the probability or consequences of an accident previously evaluated in evident.

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The relocation of the boronometer's sealed source to safe storage has not resulted in any new or different kind of accident from any previously evaluated. The proposed deletion of Specification 4.7.9.1.2.d furthermore does not remove all controls from the subject source. While maintained in storage, the requirements of Specification 4.7.9.1.2.b will govern testing of the sealed source should it be placed in service or transferred to another licensee in the future.

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

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety

The relocation of the boronometer's sealed source to safe storage does not impact the margin to safety. Controls are currently established governing sources that are stored and not in use. Therefore, deleting the current requirements of Specification 4.7.9.1.2.d does not result in a reduction in the margin of safety. Furthermore, deletion of this surveillance requirement will act to reduce radiological exposure to personnel that would normally be assigned to perform this activity.

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

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 27, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.4.9.1.2 and delete TS Table 4.4–5 to remove from the TSs the schedule for the withdrawal of reactor vessel material surveillance specimens, pursuant to the guidance provided in Generic Letter 91–01, ''Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens From Technical Specifications.'' Changes to the related Bases are also proposed.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because the accident conditions and assumptions are not affected by the proposed Technical Specification (TS) change. The Reactor Vessel Material Surveillance Program ensures the availability of data to update the in-service operating temperature and pressure limits as well as the Low Temperature Overpressure (LTOP) and Pressurized Thermal Shock (PTS) analyses. The schedule identifying the withdrawal of the surveillance specimens will be removed from the TSs; however, the proposed TS 4.4.9.1.2 will continue to require that the specimens be removed and examined to determine the changes in their material properties, as required by Appendix H to 10CFR50. The proposed surveillance specimen removal schedule conforms to ASTM [American Society for Testing and Materials] E185–82, ''Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels'' as referenced by 10CFR50, Appendix H. No changes to the design of the facility have been made. No new equipment has been added or removed and no operational setpoints have been altered.

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not add or modify any equipment nor does the proposed change involve any operational changes to any plant systems or Limiting Conditions for Operation (LCO). As required by Appendix H, the proposed change will continue to require the specimens be removed and examined to determine changes in their material properties. This change does not introduce any new accident or malfunction mechanism nor is any physical plant change required.

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

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety

Removal of the schedule from Technical Specifications is an administrative change and will have no impact on the margin of safety. Since changes to the reactor vessel material surveillance specimens withdrawal schedule are controlled by the requirements of Appendix H to 10CFR50, removing the schedule from Technical Specifications will not result in any loss of regulatory control. In addition, to ensure the surveillance specimens are withdrawn at a proper time, surveillance requirement 4.4.9.1.2 will continue to require specimens be removed and examined per the ANO–2 [Arkansas Nuclear One, Unit 2] Safety Analysis Report to determine changes in their material properties, as required by Appendix H.

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

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

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

Attorney for licensee: Nicholas S. Reynolds, Winston and 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: January 12, 2000 (NPF–38–226).

Description of amendment request: The proposed change modifies Technical Specifications (TS) 3.9.4, ''Containment Building Penetrations,'' to allow the containment equipment door, airlocks, and other penetrations to remain open, but capable of being closed, during core alterations or movement of irradiated fuel in containment. Additionally, a note, Bases changes, and Surveillance Requirements changes provide further enhancements to clarify equipment door, airlock, and penetration closure capability.

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

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

Response: The proposed change would allow the containment equipment hatch door, personnel air lock (PAL) doors, emergency air lock (EAL) doors and penetrations to remain open during fuel movement and core alterations. These penetrations are normally closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident (FHA) inside the containment. These penetrations are not initiators of any accident. The probability of a FHA is unaffected by the position of these penetrations.

The new FHA analysis with an open containment demonstrates the maximum offsite doses are well within the acceptance limits specified in SRP [Standard Review Plan] 15.7.4. This FHA analysis results in maximum offsite doses of 53.70 rem to the thyroid and 0.176 rem to the whole body. The calculated control room dose is also well within the acceptance criteria specified in GDC [General Design Criteria] 19. The analysis results in thyroid and whole body dose to the control room operator of 0.932 rem and 0.015 rem, respectively.

Therefore, the proposed change will not involve a significant increase in the probability or consequences of an 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 proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design, configuration, or method of operation of the plant beyond the standard functional capabilities of the equipment. The proposed change involves a change to the Technical Specifications (TS) that would allow the equipment hatch door, the PAL door, the EAL door and penetrations to be open during core alterations and fuel movement within the containment. Having these doors and penetrations open does not create the possibility of a new accident. Provisions to ensure the capability to close the containment will have been made in the event of a FHA.

Therefore, 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:

This proposed change has the potential for an increased dose at the site boundary due to a FHA; however, the analysis demonstrates that the resultant doses are well within the appropriate acceptance limits. The margin of safety, as defined by SRP 15.7.4, Rev. 1, has not been significantly reduced. The offsite and control room doses due to a FHA with an open containment have been evaluated with conservative assumptions, such as all airborne activity reaching the containment is released instantaneously to the outside atmosphere, will ensure the calculation bounds the expected dose. Closing the equipment hatch door and at least one door in each personnel airlock following an evacuation of the containment reduces the offsite doses in the event of a FHA and provides additional margin to the calculated offsite doses.

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, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 25, 2000.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Section 3/4.4.5, ''Reactor Coolant System—Steam Generators,'' and its associated Bases. In accordance with Framatome Technologies Incorporated Topical Report BAW–10236P, Revision 0, ''Addendum for Davis-Besse Repair Roll UTS Exclusion Zones,'' the proposed changes would modify the repair roll process to update exclusion zones and allow the use of the double repair roll for the repair of once-through steam generator tubes with defects within the upper tubesheet.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because testing and analysis have shown the proposed repair roll process to be added to Surveillance Requirement (SR) 4.4.5.4.a.7 ensures the new pressure boundary joint created by the repair roll process provides structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. The proposed repair roll process does not alter the design or operating characteristics of the steam generators or systems interfacing with the steam generators. Therefore, the proposed changes to SR 4.4.5.4.a.7 will not increase the probability of a previously evaluated accident.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed repair roll process to be added to Surveillance Requirement (SR) 4.4.5.4.a.7 ensures the new pressure boundary joint created by the repair roll process provides structural and leakage integrity equivalent to the original design and construction for all accident conditions. Should a repaired tube fail, the radiological consequences would be bounded by the existing Steam Generator Tube Rupture analysis.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase to the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because there will be no change in the operation of the steam generators or connecting systems as a result of the repair roll process added by the proposed changes to SR 4.4.5.4.a.7. The physical changes in the steam generators associated with the repair roll process have been evaluated and do not create the possibility for a new or different kind of accident from any accident previously evaluated, *i.e.,* the physical change in the steam generators is limited to the location of the primary to secondary boundary within the tubesheet. Furthermore, the repair roll process installs a pressure boundary joint equivalent to that of the original fabrication. Accordingly, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not create the possibility of any new or different kind of accident.

3. Not involve a significant reduction in a margin of safety because all of the protective boundaries of the steam generator are maintained equivalent to the original design and construction with tubes repaired by the repair roll process. Furthermore, tubes with primary system to secondary system boundary joints created by the repair roll have been shown by testing and analysis to satisfy all structural, leakage, and heat transfer requirements. The additional testing of tubes repaired by the repair roll process under existing SR 4.4.5.9 provides continuing inservice monitoring of these tubes such that inservice degradation of tubes repaired by the repair roll process will be detected. Therefore, the changes to SR 4.4.5.4.a.7 to modify the repair process do not reduce the margin of safety.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not reduce the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: November 30, 1999.

Description of amendment request: The licensee proposed to amend the unit's Technical Specifications (TS), Section 3.4.4, ''Emergency Ventilation System [EVS],'' and Section 3.4.5, ''Control Room Air Treatment System,'' to require testing consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989. The current standard specified by these sections is ANSI N510–1980. The licensee's application for amendment is a response to the NRC's Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

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 operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS change will require testing the EVS and Control Room Air Treatment System charcoal filters in accordance with ASTM D3803–1989 versus ANSI N510–1980. Neither the EVS or Control Room Air Treatment System involve initiators or precursors to an accident previously evaluated as both systems perform mitigative functions in response to an accident. Failure of either system would result in the inability to perform its mitigative function but no failure would increase the probability of an accident. Accordingly, changing the test methodology of the charcoal filters will not affect any accident precursors. Therefore, the probability of an accident previously evaluated is not increased.

The NMP1 [Nine Mile Point Unit 1] EVS is designed to limit the release of radioactive gases to the environment within the guidelines of 1OCFR1OO for analyzed accidents. The Control Room Air Treatment System is designed to limit doses to control room operators to less than the values allowed by GDC 19. Both systems contain charcoal filters which require laboratory carbon sample analysis be performed in accordance [with] ANSI [American National Standards Institute] N510–1980 as required by TS. Charcoal filter samples are tested to determine whether the filter adsorber efficiency is greater than that assumed in the design basis accident analysis. The proposed TS changes to test the charcoal material in accordance with ASTM D3803–1989 (versus ANSI N510) will assure the ability of the subject systems to perform their intended function by providing a more realistic prediction of the capability of the charcoal filters. Therefore, the proposed changes will not involve a significant increase in the consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1, 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 TS change will require testing the EVS and Control Room Air Treatment System charcoal filters in accordance with ASTM D3803–1989 versus ANSI N510–1980. This change will not involve placing these systems in new configurations or operating the systems in a different manner that could result in a new or different kind of accident. Testing in accordance with the ASTM D3803–1989 standard will assure the ability of the subject systems to perform their intended function by providing a more realistic prediction of the capability of the charcoal filters. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed TS changes will not adversely affect the performance characteristics of the EVS or Control Room Air Treatment System nor will it affect the ability of these systems to perform their intended functions. Charcoal filter samples are tested to determine whether the filter absorber efficiency is greater than that assumed in the design basis accident analysis. The proposed TS changes to test the charcoal material in accordance with ASTM D3803–1989 (versus ANSI N510–1980) will assure the ability of the subject systems to perform their intended function by providing a more realistic prediction of the capability of the charcoal filters. Also, the proposed changes are consistent with the changes recommended in NRC GL 99–02. 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: Mark J. Wetterhahn, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Acting Section Chief: Marsha Gamberoni.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment requests: November 10, 1999.

Description of amendment requests: The proposed amendments would modify Technical Specification (TS) 4.12, ''Steam Generator Tube Surveillance,'' to revise the elevated F–Star (EF*) distance from 1.62 inches to 1.67 inches based on Westinghouse Topical Report WCAP–14225, Revision 2, entitled ''F* and Elevated F* Tube Plugging Criteria for Tube with Degradation in the Tubesheet Region of the Prairie Island Units 1 and 2 Steam Generators.'' The change was necessitated by a correction of a minor error in the tubesheet bending calculation associated with the previously approved EF* criterion.

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

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

The proposed change to the EF* Distance ensures the roll expansion is sufficient to preclude tube pullout from tube degradation located below the EF* Distance, regardless of the extent of the tube degradation. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. Tube rupture and pullout is not expected for tubes using either the proposed or current EF* Distance because, in practice, the roll expanded region exceeds both distances. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is still fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island USAR [Updated Safety Analysis Report].

Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the EF* distance, any postulated faulted condition primary to secondary leakage from EF* tubes would be insignificant. Leakage testing was previously reported for 2 inch effective length hard rolls.

Thus, neither the probability nor consequences of previously evaluated accidents are affected by the proposed increase in the EF* Distance.

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

Implementation of the proposed EF* Distance does not introduce any significant changes to the plant design basis, nor does it change the way any system, structure, or component is operated. Use of EF* (either using the existing or proposed EF* Distance) does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis.

Thus, no new or different kind of accident is created by the proposed increase in EF* Distance.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety.

The proposed increase in EF* Distance will not decrease the integrity of the reactor coolant system boundary. The use of the EF* criterion has been previously demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg. Guide 1.121 (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation of the EF* criterion is any degradation indication in the tubesheet region, more than the EF* Distance below the bottom of the transition between the roll expansion and the unexpanded tube. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code used in steam generator design.

The EF* Distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analyses.

Revision of the EF* length does not affect the integrity of the existing EF* tubes which are in service due to the conservative length of the additional reroll.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: January 5, 2000.

Description of amendment request: This request proposes to change Technical Specification Section 3/4 6.1.6, including its Bases, and to add Section 6.8.4.h. The proposed changes support the new requirements of 10 CFR 50.55a, which require licensees to update their Containment Vessel Structural Integrity Programs to incorporate the provisions of ASME Section XI, Subsection IWL (1992 Edition with 1992 Addenda) and the five additional provisions found in 10 CFR 50.55a(b)(2)(viii).

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

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

The proposed changes revise the surveillance requirements for containment reinforced concrete and unbonded post-tensioning systems inservice examinations as required by 10 CFR 50.55a(b)(2)(vi) and 10 CFR 50.55a(b)(2)(viii). The revised requirements affect the inservice inspection program designed to detect structural degradation of the containment reinforced concrete and unbonded post-tensioning systems and do not affect the function of the containment reinforced concrete and unbonded post-tensioning system components. The reinforced concrete and unbonded post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility.

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

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

The proposed changes do not involve a modification to the physical configuration of the plant (*i.e.,* no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new of different requirements or introduce a new accident initiator, accident precursor or malfunction mechanism. The proposed changes provide an NRC approved ASME Code inspection/testing methodology to assure age related degradation of the containment structure will not go undetected. The function of the containment reinforced concrete and unbonded post-tensioning system components are not altered by this change. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different type than previously evaluated.

3. This proposed change does not involve a significant reduction in a margin [of] safety.

The Reactor Building internal design pressure is 57 psig and the maximum peak pressure from a postulated steam line break is 53.5 psig. The proposed change does not impact the margin of safety included in the design pressure compared to the peak calculated pressure because the proposed activity does not alter, in any way, the available force provided by the tendons. Additionally, the proposed activity does not affect the initial temperature conditions within the Reactor Building assumed in the accident analysis for a steam line break. Therefore, 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 amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: January 27, 2000.

Description of amendment request: The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS), Section 5.6.1, are being revised to replace the maximum reference fuel assembly K infinity (K_∞) with a figure of Integral Fuel Burnable Absorbers (IFBA) rods per assembly versus nominal fuel enrichment. This change will assure that the reactivity requirements for spent fuel storage remain satisfied. Additionally, the requirement for new fuel storage is being revised to remove K_∞ since IFBAs are not considered or required in the criticality analysis for new fuel storage.

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

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

The proposed changes revise the methodology utilized in determining the IFBA requirement for storage of spent fuel. IFBA credit is not used in the new fuel storage criticality analysis performed by Westinghouse. Removing K infinity (K_∞) from these Specifications and replacing the spent fuel requirement with the IFBA-enrichment curve will not result in any increase in the probability or consequences of an accident previously evaluated. The analysis of concern is the criticality analysis for storage of fuel in the spent fuel storage racks. The analysis must conclude that fuel stored in the configurations allowed in the spent fuel storage racks will not result in any unplanned criticality.

The IFBA rods per assembly versus the nominal enrichment of the fuel assembly curve and the K_∞ methodology were both developed to ensure that K_{eff} in the spent fuel storage racks remains less than or equal to 0.95 under all postulated conditions. This limit is included in the VCSNS licensing basis. The IFBA versus enrichment curve results in determining more accurate IFBA requirements than the K_∞ methodology, and continues to maintain the licensing basis limit.

This change will not revise the geometry of the spent fuel storage racks, the poisons present to prevent criticality, or coolant capabilities. The licensing basis limit for reactivity control of the spent fuel storage racks remains satisfied.

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

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

The proposed change does not result in any change to the design or operation of the spent fuel pool or any support systems associated with the spent fuel pool. The IFBA requirements developed from using the IFBA versus enrichment curve are potentially more conservative than developed using the K_∞ methodology. There are no scenarios that are postulated to occur that would create the possibility of a new or different kind of accident from any previously evaluated in the FSAR (see original) or FPER (see original).

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. IFBA is not assumed in any criticality analysis performed for new fuel storage. This change incorporates a more accurate method for determining IFBA requirements for fuel storage in the spent fuel storage racks. Both the current methodology and the proposed methodology have been reviewed and approved by the NRC in WCAP–14416–NP–A as acceptable methods for assuring that the licensing basis for the spent fuel pool reactivity limit remain satisfied. Therefore, the margin of safety with respect to unplanned criticality, for the storage of fuel in the spent fuel storage racks is not reduced.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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: December 17, 1999 (TS 99–25).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Operating Licenses DPR–77 (Unit 1) and DPR–79 (Unit 2) by modifying License Provision Statement 2.B.(5), in conjunction with an exemption to 10 CFR 50.54(ee), to allow temporary storage of low-level radioactive waste generated at the Watts Bar Nuclear Plant (WBN) at the SQN plant site.

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

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

The probability of occurrence or the consequences for an accident or malfunction is not increased. Design basis accidents were previously analyzed by TVA and reviewed by NRC as part of the materials license process for the on-site storage facility (OSF). The intended future usage of the OSF is bounded by those analyses, with the exception of transport from WBN to SQN. Transport from WBN to SQN involves a distance of only 35 miles, which is very likely a small increment of the distance to any final off-site repository. For example, the 35-mile transit from WBN to SQN is much less than the 370-mile distance from WBN to Barnwell, South Carolina. The shipment of LLRW from WBN was reviewed as part of the WBN Unit 1 operating license request (WBN Final Safety Analysis Report [FSAR] Section 11.5.6). As with any shipment of low-level radioactive waste (LLRW), all Department of Transportation (DOT) requirements will be met.

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

A possibility for an accident or malfunction of a different type than any evaluated previously in SQN's FSAR is not created by the proposed change; nor is the possibility for an accident or malfunction of a different type. Potential accidents were previously analyzed by TVA and reviewed by NRC as part of the materials license process for the OSF. The intended future usage of the OSF is bounded by those analyses, with the exception of transport from WBN to SQN. Radwaste shipments from WBN to SQN will be no different than any other radwaste shipment except that the distance is only 35 miles. This transportation route does not present any significant potential negative impacts on the public health and safety. As with any shipment of LLRW, all DOT requirements will be met.

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

The proposed amendment will not involve a significant reduction in the margin of safety. The margin of safety was previously analyzed by TVA and reviewed by NRC as part of the materials license process for the OSF. The intended future usage of the OSF is bounded by those analyses, with the exception of transport from WBN to SQN. The transport route from WBN to SQN, which involves a distance of only 35 miles, does not present any significant potential negative impacts on the public health and safety [and] is very likely a small increment of the distance to any final off-site repository. For example, this is much less than the distance to Barnwell. The shipment of LLRW from WBN was reviewed as part of the WBN Unit 1 operating license request (WBN FSAR Section 11.5.6). As with any shipment of LLRW, all DOT requirements will be met.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: January 13, 2000 (Reference Number TXX–00011).

Brief description of amendments: The proposed amendments would change the Comanche Peak Steam Electric Station (CPSES) Technical Specification (TS) as follows: (1) Revise TS 3.8.3 (Condition B and Surveillance Requirement (SR) 3.8.3.2) to conservatively increase the required emergency diesel generator (DG) lube oil inventory values, (2) revise TS SR 3.8.3.2 to add a note stating that the surveillance is not required to be performed until the diesel has been in shutdown greater than 10 hours, and (3) delete the footnote associated with SR 3.8.4.7.

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

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

(a) The proposed changes establish more conservative DG lube oil inventory levels to support required DG operations. Conservatively revising the required lube oil levels does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(b) The proposed change to add a surveillance note cannot affect the probability or consequences of any accident. When surveillances are done, it cannot initiate an accident or affect the course of mitigation. Lube oil levels are checked after each run. If the lube oil level was at the minimum required ''1.75 inches below the low static level'' at the start of a normal 24 hour surveillance run, 5 days of lube oil inventory is provided above the Condition B level of ''5.5 inches below the low static level.'' Allowing 10 hours after the surveillance run to check the static level is not significant because relative lube oil level is maintained during engine run through the use of an indicator on the panel ensuring adequate oil level during and just after the run. The Condition B lube oil inventory ensures a minimum of [2] days of operation before any addition of lube oil would be needed. In the event of an accident which requires extended run of the emergency diesel generators, lube oil can be added with the engines running.

(c) Deletion of the footnote associated with SR 3.8.4.7, which provided a one time exception for the battery surveillance, is an administrative change and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

(a) Plant procedures are only altered to the extent that the revised specification will enhance the monitoring of the DG lube oil inventory level to support required DG operation at full load conditions. These changes ensure continued support of the safety related DG, do not involve any physical alteration to the plant, and do not affect their failure or failure modes.

(b) The proposed change to add a surveillance note [does] not involve any physical alteration to the plant and [does] not affect their failure or failure modes.

(c) Deletion of the footnote associated with SR 3.8.4.7, which provided a one time exception for the battery surveillance, is an administrative change and will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

(a) The proposed changes will not alter any accident analysis assumptions, initial conditions, or results. Conservatively revising the required DG lube oil levels will ensure proper DG operations as assumed in the safety analyses.

(b) The proposed change to add a note will not alter any accident analysis assumptions, initial conditions, or results. Conservatively revising the required conditions for DG lube oil level surveillance will ensure proper DG operations as assumed in the safety analyses.

(c) Deletion of the footnote associated with SR 3.8.4.7, which provided a one time exception for the battery surveillance, is an administrative change and does not involve a significant reduction in a margin of safety.

Therefore, these 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: George L. Edgar, Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 13, 2000 (Reference Number TXX–00010).

Brief description of amendments: The proposed amendments would change Comanche Peak Steam Electric Station (CPSES) Technical Specification Surveillance Requirement (SR) 3.3.1.10 to add Note 3 which would allow entry into Modes 2 or 1 without the performance of N–16 detector plateau verification until 72 hours after achieving equilibrium conditions at greater than or equal to 90% of rated thermal power.

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

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

The proposed change is considered to be a correction of an editorial error. The proposed revision to SR 3.3.1.10 is consistent with the current CPSES licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is considered to be an editorial correction and does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change is considered to be an editorial correction and does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: George L. Edgar, Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of application request: January 14, 2000 (ULNRC–04172).

Description of amendment request: The proposed amendment would revise several sections of the improved Technical Specification (ITSs) to correct eight editorial errors made in either (1) The application dated May 15, 1997, (and supplementary letters) for the ITSs or (2) the certified copy of the ITSs that was submitted in the licensee's letters of May 27 and 28, 1999. The ITSs were issued as Amendment No. 133 by the staff in its letter of May 28, 1999, and will be implemented by the licensee to replace the current TSs by April 30, 2000. There are no changes in any requirements in the ITSs. The proposed changes to the ITSs are:

(1) The correct abbreviation in the table of contents, ITS page 2, Section 3.3.7, is ''CREVS'' instead of ''CREFS''.

(2) The Condition D for limiting condition for operation (LCO) 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' has a reference to itself (Condition D) that should be deleted on ITS page 3.7–5.

(3) The spelling of ''required'' will be corrected in the definition of the Term Actions on ITS page 1.1–1.

(4) The completion time of 8 hours for Required Action A.2 of Example 1.3–6 on ITS page 1.3–10 will be properly relocated to be on the same line as A.2.

(5) The note for Condition D of LCO 3.7.4, ''Atmospheric Steam Dump Valves (ASDs),'' on ITS page 3.7–10 will be made the full column width of the required action column.

(6) The word boundary in the note for LCO 3.7.13, ''Emergency Exhaust System (EES),'' on ITS page 3.7–31, will not be capitalized.

(7) The note for Condition A of LCO 3.7.16, ''Fuel Storage Pool Boron Concentration,'' on ITS page 3.7–36 will be made the full column width of the required action column.

(8) The colon in 3.1:5 will be replaced by a period to have 3.1.5 in the list of specifications given in item a.7 of Section 5.6.5, ''Core Operating Limits Report (COLR),'' on ITS page 5.0–29.

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

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

The proposed changes involve corrections to the ITS that are associated with the original conversion application and supplements or the certified copy of [the] ITS. The changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements of the Technical Specifications. As such, the administrative changes do not effect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose any new or eliminate any old requirements.

Thus, the changes do 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 will not reduce a margin of safety because they have no effect on any safety analyses assumptions. The changes are administrative in nature.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 21, 1999.

Description of amendment request: The proposed amendment would change Section 15.3 of the Technical Specifications in order to more clearly define the requirements for the service water (SW) system operability. The December 21, 1999, application supercedes the July 30, 1998, application that was previously noticed in the **Federal Register** (63 FR 71976) on December 30, 1998.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The Service Water System is primarily a support systems required to be operable for accident mitigation. Portions of the SW system supplying the containment fan coolers also function as part of the containment pressure boundary under post accident conditions. Failures within the SW system are not an initiating condition for any analyzed accident.

Analyses performed demonstrate that under the Technical Specifications allowable configurations, the SW system will continue to perform all required functions. The SW system is capable of supplying the required cooling water flow to systems required for accident mitigation. That is, the SW system removes the required heat from the containment fan coolers and residual heat removal heat exchangers ensuring containment pressure and temperature profiles following an accident are as evaluated in the FSAR [Final Safety Analysis Report]. This in turn ensures that environmental qualification of equipment inside containment is maintained and thus functions as required post-accident.

SW system response post accident is within all design limits for the system. Transient and steady state forces within the system remain within all design and operability limits, thereby maintaining the integrity of the system inside containment and the integrity of the containment pressure boundary. Assumptions dependent on the containment pressure profile for containment leakage assumed in the radiological consequences analyses remain valid.

In addition, removing required heat from containment ensures that cooling of the reactor core is accomplished for long-term accident mitigation.

Therefore, operation of the SW system as proposed will not result in a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes do not alter the way in which the SW system performs its design functions nor the design criteria of the system. The proposed changes do not introduce any new or different normal operation or accident mitigation functions for the system. Therefore, no new accident initiators are introduced by the proposed changes. Operation of [the] SW system as proposed cannot result in a new or different kind of accident from any accident previously evaluated.

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

Analyses performed in support of the proposed amendments demonstrate that the SW system continues to perform its function as assumed and credited in the accident analyses and radiological consequence analyses performed for the Point Beach Nuclear Plant. The SW flow analyses conservatively assume limiting calculational parameters such as minimum allowed IST [inservice testing] pump performance curves, minimum credible pump bay level, maximum postulated lake temperature, inclusion of system water leakage, maximum flow through system temperature control valves, bounding values for system throttle valve settings and impacts of instrument inaccuracy. Therefore, the analyses and results are not changed. All analysis limits for the system remain met. The SW system continues to be operated and responds within all design limits for the system. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments cannot result in a significant reduction in 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 H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Claudia M. Craig.

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.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of application for amendment: December 28, 1999.

Brief description of amendment: The amendment would revise the reactor vessel material coupon withdrawal schedule specified in Technical Specifications Table 4.4.6.1.3–1, entitled ''Reactor Vessel Material Surveillance Program-Withdrawal Schedule.''

Date of publication of individual notice in **Federal Register:** January 14, 2000 (65 FR 2443).

Expiration date of individual notice: February 14, 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).

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

Date of application for amendments: February 26, 1999, as supplemented May 21, 1999.

Brief description of amendments: The amendments revise the Technical Specifications to extend the completion time for one inoperable low pressure safety injection subsystem from 72 hours to 7 days. These amendments provide partial response to the licensee's application for amendments. The remaining request will be addressed under separate correspondence.

Date of issuance: February 1, 2000.

Effective date: February 1, 2000, to be implemented within 45 days.

Amendment Nos.: Unit 1—124, Unit 2—124, Unit 3—124.

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

*Date of initial notice in **Federal Register:*** April 7, 1999 (64 FR 17023).

The May 21, 1999, supplement provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: August 27, 1999, as supplemented September 20, 1999.

Brief description of amendments: The amendments would modify the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Technical Specifications to allow placement of one or more assemblies on spent fuel rack spacers to support fuel reconstitution activities in the spent fuel pool.

Date of issuance: February 3, 2000.

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

Amendment Nos.: 233 and 209.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 22, 1999 (64 FR 51345).

The September 20, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated February 3, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 4, 1999, as supplemented December 3, 1999, and January 11, 2000.

Brief description of amendment: This amendment revises Technical Specification 6.9.1.6.2 to incorporate analytical methodology references which are used to determine core operating limits. The analytical methodologies referenced are documented in topical reports which have been accepted by the Nuclear Regulatory Commission for referencing in licensing applications.

Date of issuance: February 10, 2000.

Effective date: February 10, 2000.

Amendment No.: 94.

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

*Date of initial notice in **Federal Register:*** August 25, 1999 (64 FR 46426).

The December 3, 1999, and January 11, 2000, submittals 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 February 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 12, 1999, as supplemented by letter dated January 10, 2000.

Brief description of amendments: The amendments changed Technical Specification (TS) 3/4.6.K to revise the reactor pressure boundary pressure-temperature limits, changed TS 3/4.12.C to delete a special test exception which allows performance of the hydrostatic test above 212 degrees Fahrenheit while in Mode 4, and changed TS 3/4.6.P to clarify the operability requirements for the residual heat removal system during the hydrostatic test.

Date of issuance: February 4, 2000.

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

Amendment Nos.: 195 & 191.

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

*Date of initial notice in **Federal Register:*** December 15, 1999 (64 FR 70081).

The January 10, 2000, letter did not change the original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 4, 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 October 20, 1999.

Brief description of amendment: The amendment consists of changes to Surveillance Requirements (SR) 3.8.4.6 of Technical Specifications (TS) 3.8.4, ''DC Sources—Operating'' and SR 3.8.5.1 of TS 3.85, ''DC Sources—Shutdown.''

Date of issuance: January 28, 2000.

Effective date: January 28, 2000, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 160.

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

*Date of initial notice in **Federal Register:*** August 25, 1999 (64 FR 46432).

The October 20, 1999, supplemental letter corrected the page numbering of the technical specifications 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 January 28, 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: January 25, 1999, as supplemented by letter dated December 9, 1999.

Brief description of amendment: The amendment consists of a modification to TS 3/4.5.1 to allow up to 72 hours to restore safety injection tank (SIT) operability if one SIT is inoperable due to boron concentration not within the limits or the inability to verify level or pressure. The proposed change also allows up to 24 hours to restore SIT operability if one SIT is inoperable due to other reasons when reactor coolant system pressure is greater than or equal to 1750 pounds per square inch, absolute.

Date of issuance: February 7, 2000.

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

Amendment No.: 155.

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

Date of initial notice in ***Federal Register:*** February 24, 1999 (64 FR 9191).

The December 9, 1999, letter provided additional information that did not change the scope of the application as initially noticed or change proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 7, 2000.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: October 16, 1998, as supplemented by letter dated May 10, 1999, and December 8, 1999.

Brief description of amendment: This amendment changes portions of the Technical Specifications regarding the Service Water System.

Date of issuance: February 3, 2000.

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

Amendment No.: 89.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 2, 1998 (63 FR 66596).

The May 10 and December 8, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 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: July 16, 1999.

Brief description of amendment: The amendment relocates Technical Specification (TSs) 3/4.9.3.2, ''Refueling Operations, Spent Fuel Temperature,'' 3/4.9.3.3, ''Refueling Operations, Decay Time,'' 3/4.9.5, ''Refueling Operations, Communications,'' 3/4.9.6, ''Refueling Operations, Crane Operability—Containment Building,'' and 3/4.9.7, ''Refueling Operations, Crane Travel—Spent Fuel Storage Building,'' to the Millstone, Unit No. 2 Technical Requirements Manual. The associated Bases pages and index pages are also modified to address the proposed change.

Date of issuance: February 10, 2000.

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

Amendment No.: 240.

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

Date of initial notice in ***Federal Register:*** October 6, 1999 (64 FR 54378).

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

No significant hazards consideration comments received: No.

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

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

Brief description of amendment: The amendment implements the Radioactive Effluent Technical Specifications and makes changes necessary to implement the revised 10 CFR Part 20.

Date of issuance: February 7, 2000.

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

Amendment No.: 199.

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

Date of initial notice in ***Federal Register:*** August 25, 1999 (64 FR 46442).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 7, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 16, 1998, as supplemented January 28, 1999.

Brief description of amendment: The amendment relocates the Chemical and Volume Control System Technical Specifications.

Date of issuance: February 7, 2000.

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

Amendment No.: 200.

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

Date of initial notice in ***Federal Register:*** February 24, 1999 (64 FR 9200).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 7, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 6, 1999.

Brief description of amendments: The amendments revised Technical Specification Definition 1.9, ''Core Alterations,'' to explicitly define core alterations as the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel.

Date of issuance: February 1, 2000.

Effective date: February 1, 2000, to be implemented within 30 days.

Amendment Nos.: Unit 1–123; Unit 2–111.

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

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73099).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 1, 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: October 12, 1999.

Brief description of amendments: These amendments revise Technical Specification Section 3.9.4.c, ''Containment Building Penetrations,'' and the associated bases to allow use of administrative controls to unisolate certain containment penetrations during refueling operations.

Date of issuance: February 11, 2000.

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

Amendment Nos.: 249 and 240.

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

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1928).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 11, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 22, 1999, as supplemented December 17, 1999.

Brief description of amendments: The amendments reflect changes to the Technical Specifications in order to incorporate the Westinghouse 422V+ fuel assemblies into the reactor cores.

Date of issuance: February 8, 2000.

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

Amendment Nos.: 193 and 198.

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

Date of initial notice in ***Federal Register:*** July 28, 1999 (64 FR 40910).

The December 17, 1999, letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not affect the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 16th day of February, 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–4236 Filed 2–22–00; 8:45 am]

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OFFICE OF MANAGEMENT AND BUDGET

Budget Rescissions and Deferrals

TO THE CONGRESS OF THE UNITED STATES:

In accordance with the Congressional Budget and Impoundment Control Act of 1974, I herewith report three rescissions of budget authority, totaling $128 million, and two deferrals of budget authority, totaling $1.6 million.

The proposed rescissions affect the programs of the Department of Energy and the Department of Housing and Urban Development. The proposed deferrals affect programs of the Department of State and International Assistance Programs.

William J. Clinton

THE WHITE HOUSE,

February 9, 2000.

BILLING CODE 3110–01–P