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 November 17, 2000, through December 1, 2000. The last biweekly notice was published on November 29, 2000.

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative 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's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 12, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly 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). 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly 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).

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: July 5, 2000.

Description of amendment request: The proposed amendment would revise the maximum power level specified in each unit's license; revise the value of rated thermal power of each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt and the reference source for conversion factors in the calculation of Dose Equivalent Iodine 131 in the technical specification (TS) definitions; add a Departure from Nucleate Boiling Ratio (DNBR) limit specifically for a thimble cell; increase the minimum limit for reactor coolant system (RCS) total flow; revise the steam generator laser welded sleeve plugging limit; and reduce the peak calculated containment internal pressure P_a for the design basis loss-of-coolant accident (LOCA).

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?

A. Evaluation of the Probability of Previously Evaluated Accidents.

Plant systems and components have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, some components will be modified prior to implementation of uprated power operations to accommodate the revised operating conditions. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current TS surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated in accordance with current design requirements under uprated conditions, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Trip or Engineered Safety Features (ESF) setpoints.

B. Evaluation of the Consequences of Previously Evaluated Accidents.

The radiological consequences were reviewed for all design basis accidents (DBAs) (*i.e.*, both Loss of Coolant Accident (LOCA) and non-LOCA accidents) previously analyzed in the UFSAR. The analysis showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remain either unchanged or have not significantly increased due to operation at uprated power conditions. The radiological consequences of all DBAs continue to meet established regulatory limits.

The proposed addition of Table E–7 of NRC Regulatory Guide 1.109, ''Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,'' Revision 1, 1977, or International Commission on Radiological Protection (ICRP) 30, ''Limits for Intakes of Radionuclides by Workers,'' Supplement to Part 1, page 192–212, Table titled, ''Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity,'' for thyroid dose conversion factors, will not significantly increase the consequences of an accident previously evaluated. If Regulatory Guide 1.109, or ICRP 30, Supplement to Part 1, are used to calculate maximum dose equivalent iodine specific activity, the total RCS iodine activity may increase, depending on the iodine nuclide mix, and this activity is used to calculate the doses resulting from a Main Steam Line Break (MSLB) or other analyzed accident. The calculated thyroid doses resulting from an MSLB or other analyzed accident would not increase as the corresponding dose conversion factors would be used to calculate the offsite thyroid doses. For a given Dose Equivalent I–131 concentration in the RCS, the offsite dose predicted using the dose conversion factors in either Table E–7 of Regulatory Guide 1.109, or ICRP 30, Supplement to Part 1, is less than that predicted by Table III of Atomic Energy Commission (AEC) Technical Information Document TID–14844, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' which is currently referenced in the TS definition of Dose Equivalent I–131.

ICRP–30 is the updated reference for thyroid dose conversion factors used in the power uprate accident analysis radiological evaluation. The current version of 10 CFR 20, ''Standards for Protection Against Radiation,'' also utilizes ICRP–30 data.

Based on the analysis, it is concluded that the proposed TS changes do not involve a significant 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 configuration, operation and accident response of the Byron Station and the Braidwood Station systems, structures or components are unchanged by operation at uprated power conditions or by the associated proposed TS changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components. The basic design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any Reactor Trip or ESF actuation setpoints.

Based on this analysis, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed TS changes do not have an adverse effect on any safety-related system. Therefore, the proposed TS 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?

A comprehensive analysis was performed to support the power uprate program at the Byron Station and the Braidwood Station. This analysis identified and defined the major input parameters to the NSSS [Nuclear Steam Supply System], reviewed NSSS design transients, and reviewed the capabilities of the NSSS fluid systems, NSSS/BOP [balance-of-plant] interfaces, NSSS control systems, and NSSS and BOP components. All appropriate NSSS accident analysis was reperformed to confirm acceptable results were maintained and that the radiological consequences remained within regulatory limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analysis confirmed that all NSSS and BOP systems and components are capable, some with minor modifications, to safely support operations at uprated power conditions.

To support the operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 at uprated power conditions, nuclear fuel Departure from Nucleate Boiling Ratio (DNBR) reanalysis was required to define new core limits, axial offset limits, and Condition II, ''Faults of Moderate Frequency,'' acceptability. This analysis included review of the following events: loss of RCS flow, reactor coolant pump locked rotor, feedwater malfunction, dropped control rod, steamline break, and control rod withdrawal from a subcritical condition. DNB design criteria was met for all events.

NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1, dated April 1995, allows Dose Equivalent I–131 to be calculated using any one of three dose conversion factors; Table III of TID–14844, 1962, Table E–7 of NRC Regulatory Guide 1.109, Revision 1, 1977, or ICRP 30, Supplement to Part 1. Using thyroid dose conversion factors other than those given in TID–14844 results in lower doses and higher allowable activity but is justified by the discussion given in the **Federal Register** (*i.e.*, **Federal Register** (FR) page 23360 Vol. 56, May 21, 1991). This discussion accompanied the final rulemaking on 10 CFR 20, ''Standards for Protection Against Radiation,'' by the NRC. In that discussion, the NRC stated that it was incorporating modifications to existing concepts and recommendations of the ICRP into NRC regulations. Incorporation of the methodology of ICRP–30 into the 10 CFR 20 revision was specifically mentioned with the changes being made resulting from changes and updates in the scientific techniques and parameters used in calculating dose. This FR reference clearly shows that the NRC was updating 10 CFR 20 to incorporate ICRP–30 recommendations and data. Regulatory Guide 1.109 thyroid dose conversion factors are higher than the ICRP–30 thyroid dose conversion factors for all five iodine isotopes of concern. Therefore, using Regulatory Guide 1.109 thyroid dose conversion factors to calculate Dose Equivalent I–131 is more conservative than ICRP–30 and is therefore acceptable. For a given Dose Equivalent I–131 concentration in the Reactor Coolant, the offsite dose predicted using the dose conversion factors in either Table E–7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or ICRP 30, Supplement to Part 1, is less than that predicted by Table III of TID–14844 which is currently referenced in the TS definition of Dose Equivalent I–131.

ICRP–30 is the updated reference source used in the power uprate accident analysis radiological evaluation. All regulatory acceptance criteria continue to be met and adequate safety margin is maintained.

Revising the minimum limit for RCS total flow from greater than or equal to 371,400 gpm to greater than or equal to 380,900 gpm does not represent a significant reduction in the margin of safety. The reactor coolant pumps run at full flow and have a total flow capacity greater than 380,900 gpm. The analysis has shown that DNBR criteria has been met for all normal operational transients and loss of flow accident scenarios.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

The proposed change revises the plugging limit for laser welded sleeves from 40% to 38.7% of nominal wall thickness. The analysis performed in support of the power uprate effort, indicated that it is necessary to remove steam generator (SG) tubes with laser welded sleeves from service upon discovering an imperfection depth of 38.7% wall thickness to ensure the structural integrity of SG tubes which have been sleeved thereby precluding the occurrence of an SG tube rupture of sleeved tubes under all operating conditions. The previous laser welded sleeve plugging limit was based on an analysis that used lower tolerance limit material strength values. The new analysis methodology, required for laser welded sleeves, uses minimum strength properties from the American Society of Mechanical Engineers Code. As determined by the new analysis, reducing the plugging limit from 40% to 38.7% maintains a comparable margin of safety to the previous analysis.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicated that the safety margin improved, even though the mass and energy release due to a LOCA under uprated power conditions increases. Based on new and improved analytical methodologies, P_a, the peak calculated containment internal pressure for the design basis LOCA, is 42.8 psig as compared to the current value of 47.8 psig for Unit 1; and is 38.4 psig as compared to the current value of 44.4 psig for Unit 2, for both Byron Station and Braidwood Station.

Radiological consequences of the following accidents were reviewed: Main Steamline Break, Locked Reactor Coolant Pump (RCP) Rotor, Locked RCP Rotor with Power-Operated Relief Valve Failure, Rod Ejection, Small Line Break Outside Containment, Steam Generator Tube Rupture, Large Break Loss of Coolant Accident, Small Break Loss of Coolant Accident, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents did not show a significant change due to uprated power conditions and 10 CFR 100 limits continue to be met.

The analyses supporting the power uprate program have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that the proposed TS changes do not involve a significant reduction in the margin of safety.

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

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.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant

Date of amendment request: July 7, 2000.

Description of amendment request: The proposed amendment would add a new license condition which would approve the License Termination Plan dated July 7, 2000, and allow the licensee to make changes to the approved License Termination Plan without prior Nuclear Regulatory Commission (NRC) approval if certain criteria specified in the license condition are met.

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

CYAPCO has reviewed the proposed change to the Operating License in accordance with the requirements of 10 CFR 50.92, ''Issuance of Amendment,'' and concluded that the change does not involve a significant hazards consideration (SHC). The proposed change does not involve an SHC because the change would not:

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

Currently, the bounding airborne radioactivity event given in the Haddam Neck Plant UFSAR [Updated Final Safety Analysis Report] is the resin container accident. Whereas previously doses associated with gaseous waste system accidents would have bounded those associated with solid waste system failures, the small amount of radioactivity contained within the gaseous radioactive waste system with the plant in the permanently defueled condition results in this system's failure no longer being bounding. The curie content of the resin container was based on the actual radioactivity inventory collected on the resin from the reactor coolant system decontamination. This corresponded to approximately 90% of the NRC Class C burial limits. Consistent with NUREG–0782 for a resin fire, one percent of the activity of the container was assumed to be released to the environment. The 1% bounds the potential airborne release fraction from various resin incidents, such as an exothermic reaction during dewatering, dropping of a high integrity container, or a resin spill. Other airborne particulate radwaste or radioactive material accidents considered in the UFSAR but bounded by the resin container fire are as follows:

- a fire in the radwaste storage facility,
- a drop of a component (*e.g.*, steam generator, reactor vessel, or heat exchanger) being removed from the site,
- a van of radioactive waste materials consumed by a fire while stored in the yard area on-site,
- a radiological HEPA [High-Efficiency Particulate Air] filter rupture,
- segmentation of components or structures during loss of local engineering controls,
- an oxyacetylene tank explosion, or
- an explosion of liquid propane gas leaked from a front-end loader.

The UFSAR also discusses a fuel handling accident in the fuel building, involving the drop of a spent fuel assembly onto the fuel racks. The postulated drop assumes the rupture of all fuel rods in the associated assembly. The probability or consequences of this accident would not be increased during any future fuel transfer operations in the spent fuel pool related to decommissioning. Transfer of the spent fuel to canisters for dry cask storage will involve additional restrictions contained in the cask certificate of compliance in order to maintain decommissioning activities within the assumption and consequences of the fuel handling accident.

The requested license amendment is consistent with plant activities described in the Post Shutdown Decommissioning Activities Report (PSDAR) and the HNP [Haddam Neck Plant] Decommissioning UFSAR. Accordingly, no systems, structures, or components that could initiate the previously evaluated accidents or are required to mitigate these accident are adversely affected by this proposed change. Therefore, the proposed change does not involve an increase in the probability or consequences of any previously evaluated accident.

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

Accident analyses related to decommissioning activities are addressed in the UFSAR. The requested license amendment is consistent with the plant activities described in the HNP Decommissioning UFSAR and the PSDAR. Thus, the proposed change does not affect plant systems, structures, or components in a way not previously evaluated. No new failure mechanisms will be created by this activity, and the proposed activity does not create the possibility of a new or different kind of accident than those previously evaluated.

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

The License Termination Plan (LTP) is a plan for demonstrating compliance with the radiological criteria for license termination as provided in 10 CFR 20.1402. The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use (one of the criteria of 10 CFR 20.1402). This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Since the License Termination Plan was designed to comply with the radiological criteria for license termination for unrestricted use, the LTP supports this margin of safety.

In addition, the LTP provides the methodologies and criteria that will be used to perform remediation activities of residual radioactivity to demonstrate compliance with the ALARA [as low as reasonably achievable] criterion of 10 CFR 20.1402.

Additionally, the LTP was designed with recognition that (a) the methods in MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) and (b) the building surface contamination levels are not directly applicable to use with complex nonstructural components. Therefore, the LTP states that nonstructural components remaining in buildings (*e.g.*, pumps, heat exchangers, etc.) will be evaluated against the criteria of RG [Regulatory Guide] 1.86 to determine if the components can be released for unrestricted use. The LTP also states that materials, surveyed and evaluated as a part of normal decommissioning activities and prior to implementation of the final status survey, will be surveyed for release using current site procedures to demonstrate compliance with the ''no detectable'' criteria. Such materials that do not pass these criteria will be controlled as contaminated.

Also, as previously discussed, the bounding accident for decommissioning is the resin container accident. Since the bounding decommissioning accident results in more airborne radioactivity than can be released from other decommissioning events, the margin of safety associated with the consequences of decommissioning accidents is not reduced by this activity.

Thus, 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: Mr. Robert K. Gad, III, Ropes & Gray, One International Plaza, Boston, Massachusetts 02110–2624.

NRC Section Chief: Michael T. Masnik.

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

Date of amendment request: November 2, 2000.

Description of amendment request: The proposed amendment would revise the WNP–2 Technical Specifications (TS) to incorporate long-term power stability solution requirements. The proposed changes reflect: (1) The addition of a new TS Section 3.3.1.3, ''Oscillation Power Range Monitoring (OPRM) Instrumentation,'' (2) a revision to TS Section 3.4.1, ''Recirculation Loops Operating,'' to remove monitoring specifications that would no longer be necessary upon activation of the automatic OPRM instrumentation, and (3) a revision to TS 5.6.5 to include in the Core Operating Limits Report (COLR) the applicable operating limits for the OPRMs, and also reference the topical report which describes the analytical methods used to determine the setpoint values for the OPRM.

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

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements for the OPRM system and allows operation in regions of the power-to-flow map currently restricted by the requirements of interim corrective actions (ICAs) and certain limiting conditions of operation of Technical Specification 3.4.1. The restrictions of the ICAs and Technical Specification 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of thermal-hydraulic oscillations that may develop into a thermal-hydraulic instability event. A thermal-hydraulic instability event has the potential to challenge the minimum critical power ratio (MCPR) safety limit. The OPRM system can automatically detect and suppress conditions necessary for thermal-hydraulic instability. With the installation of the OPRM system, the restrictions of the ICAs and Technical Specification 3.4.1 are no longer required to prevent a potential challenge to the MCPR safety limit during an anticipated instability event.

The probability of a thermal-hydraulic event is dependent on power-to-flow conditions such that only during operation inside specific regions of the power-to-flow map, in combination with power shape and inlet enthalpy conditions, can the occurrence of an instability event be postulated to occur. Operation in these regions may increase the probability that operation with conditions necessary for a thermal-hydraulic instability can occur. When the OPRM system is operable, conditions consistent with the imminent onset of oscillations are automatically detected and the conditions necessary for oscillations are suppressed, which decreases the probability of an instability event. In the event the trip capability of the OPRM is not maintained, the proposed change limits the period of time before an alternate method to detect and suppress thermal-hydraulic oscillations is required. The probability of a thermal-hydraulic instability event may be increased during the limited period of time that operation is allowed at conditions otherwise requiring the trip capability of the OPRM to be maintained. However, since the duration of this period of time is limited, the increase in the probability of a thermal-hydraulic instability event is not significant.

The proposed change requires the OPRM system to be operable and, thereby, ensures mitigation of thermal-hydraulic instability events with a potential to challenge the MCPR safety limit when initiated from anticipated conditions, by detection of the onset of oscillations and actuation of an RPS [reactor protection system] trip signal. The OPRM also provides the capability of an RPS trip being generated for thermal-hydraulic instability events initiated from unanticipated, but postulated conditions. These mitigating capabilities of the OPRM system will become available as a result of the proposed change and have the potential to reduce the consequences of anticipated and postulated thermal-hydraulic instability events. The OPRM installation has been evaluated and does not alter the function or capability of any other installed equipment such as the average power range monitoring (APRM) system or the RPS to mitigate the consequences of postulated events.

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

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power-to-flow map currently restricted by the requirements of ICAs and Technical Specification 3.4.1. The OPRM system uses input signals shared with APRM and rod block functions to monitor core conditions and generate an RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident.

Operation in regions currently restricted by the requirements of ICAs and Technical Specification 3.4.1 is within the nominal operating domain and ranges of plant systems and components, and within the range for which postulated equipment and accidents have been previously evaluated.

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

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

The proposed change specifies limiting conditions for operation, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power-to-flow map currently restricted by the requirements of ICAs and Technical Specification 3.4.1.

The OPRM system monitors small groups of LPRM [local power range monitor] signals for indication of local variations of core power consistent with thermal-hydraulic oscillations and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated thermal-hydraulic instability event has the potential to result in a challenge to the MCPR [minimum critical power ratio] safety limit. The OPRM system provides the capability to automatically detect and suppress conditions which might result in a thermal-hydraulic instability event and, thereby, maintains the margin of safety by providing automatic protection for the MCPR safety limit while significantly reducing the burden on the control room operators. In the event the trip capability of the OPRM is not maintained, the proposed change limits the period of time before an alternate method to detect and suppress thermal-hydraulic oscillation is required. The alternate method to detect and suppress oscillations would be comparable to current actions required by the interim corrective actions and no significant reduction in the margin of safety would result in the event that an unmitigated instability event occurred.

Operation in regions currently restricted by the requirements of ICAs and Technical Specification 3.4.1 is within the nominal operating domain and ranges of plant systems and components, and within the range assumed for initial conditions considered in the analysis of anticipated operational occurrences and postulated accidents.

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

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

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

NRC Section Chief: Stephen Dembek.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 29, 2000.

Description of amendment request: The proposed changes to the Arkansas Nuclear One, Unit 1 (ANO–1) Technical Specifications (TS) provide for the use of an Alternate Repair Criteria (ARC) for steam generator tubes with indications of outer diameter intergranular attack (ODIGA) within the upper tube sheet region of the once-through steam generators (OTSGs). Amendment 202 to the ANO–1 TS dated October 4, 1999, allowed the ARC for ODIGA indications only during Operating Cycle 16 at ANO–1. The proposed change would allow continued operation beyond Cycle 16 for ANO–1 with OTSG tubes that have ODIGA indications that are located in a defined area of the upper tube sheet.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the criteria in 10 CFR 50.92(c). A discussion of these criteria as they relate to this amendment request follows:

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

The purpose of the periodic surveillance performed on the OTSGs in accordance with ANO–1 Technical Specification (TS) 4.18 is to ensure that the structural integrity of this portion of the reactor coolant system will be maintained. The TS plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the TS as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a steam line or feedwater line break. The proposed TS change allows OTSG tubes with ODIGA indications contained within a defined area of the UTS [upper tube sheet] to remain in service with existing degradation exceeding the existing 40% through-wall (TW) plugging limit.

Extensive testing and plant experience has illustrated that ODIGA flaws confined to this area within the OTSG will not result in tube burst and tube leakage is unlikely. Therefore, allowing ODIGA flaws in this specific region to remain in service will not alter the conditions assumed in the current ANO–1 accident analysis for OTSG tube failures under postulated accident conditions. In addition, the condition of the OTSG tubes in this region are monitored during regular inspection intervals to assess for evidence of growth. Any growth noted will be addressed through testing and the operational assessment. Therefore, ANO–1 has determined that the identification, testing, monitoring, assessment, and corrective action programs provided in ANO [Arkansas Nuclear One] Engineering Report No. 00–R–1005–01, sufficiently supports this change request.

Application of the ODIGA alternate repair criteria will allow leaving tubes with ODIGA indications found in the defined area of the UTS in service while ensuring safe operation by monitoring and assessing the present and future conditions of the tubes. ANO–1 has operated since 1984 with ODIGA affected tubes in service with no appreciable effect on structural integrity or indications of tube leakage from ODIGA sources within the UTS. Through the inspection, testing, monitoring, and assessment program previously mentioned, and the on-line leak detection capabilities available during plant operation, continued safe operation of ANO–1 is reasonably assured.

Therefore, the application of the ODIGA alternate repair criteria 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 implementation of the ODIGA alternate repair criteria will not result in any failure mode not previously analyzed. The OTSGs are passive components. The intent of the TS surveillance requirements are being met by these proposed changes in that adequate structural and leak integrity will be maintained. Additionally, the proposed change does not introduce any new modes of plant operation.

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 application of an alternate repair criteria for ODIGA provides adequate assurance with margin that ANO–1 steam generator tubes will retain their integrity under normal and accident conditions. The structural requirements of ODIGA affected tubes have been evaluated satisfactorily and meet or exceed regulatory requirements. Leakage rates for these tubes within the defined region of the upper tubesheet are essentially zero and are reasonably assured to remain within the assumptions of the accident analysis by proper application of the ODIGA alternate repair criteria program. Assuming high differential pressures following an ATWS [Anticipated Transient Without Scram] or MSLB [Main Steam Line Break], if the ODIGA patches leak, the leakage would be less than the normal makeup capacity of the reactor coolant system. Since no appreciable impact is evidenced on the tubes structural integrity or its resulting leak rate, the margin to safety remains effectively unaltered.

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.

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: September 28, 2000.

Description of amendment request: The proposed changes to the Arkansas Nuclear One, Unit 1 (ANO–1) technical specifications revise the safety-related 4160 Volt (V) bus loss-of-voltage and 480 V bus degraded voltage relay allowable values.

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

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

The two 4160 V vital bus loss-of-voltage protection relays that are provided on each of the 4160 V safety buses act to mitigate the consequences of an accident by detecting a loss of voltage, isolating the safety buses, initiating load shedding schemes, and starting the associated emergency diesel generator (EDG). The safety function of the relays is unchanged by the proposed setpoint revisions. The revised settings for the loss-of-voltage protection relays will continue to provide the safety function with no appreciable additional time delay. The proposed time delays are within those assumed in the ANO–1 safety analyses. Additionally, the lower voltage settings will aid in preventing unnecessary isolation from the off-site power sources, which in turn will reduce the probability of a loss of off-site power to the unit due to off-site power system transients. Since the proposed change does not adversely impact the mitigating function of the relays, the consequences of an accident previously evaluated remains unchanged.

The two degraded voltage protection relays that are provided on each of the 480 V safety buses act to mitigate the consequences of an accident by detecting a sustained undervoltage condition, isolating the safety buses from offsite power, and starting the associated EDG. This safety function is unchanged by the proposed setpoint revisions. The revised settings for the degraded voltage protection relays will continue to provide the safety function of protecting the associated Class IE equipment from the effects of a low voltage condition. There is no proposed change to the existing timer setting and the time delays remain within those assumed in the ANO–1 safety analyses. Additionally, the revised allowable voltage settings will not result in any unnecessary isolation from the off-site power sources. Since the proposed change does not adversely impact the mitigating function of the relays, the consequences of an accident previously evaluated remains unchanged.

The ANO–1 technical specifications will continue to require the 4160 V bus loss-of-voltage functions and 480 V bus degraded voltage functions to be surveillance tested at their present frequency without changing the modes in which the surveillance is required or the modes of applicability for these components. The technical specifications will continue to require the same actions as currently exist for the inoperability of one or more of the 4160 V bus loss-of-voltage channels or the 480 V bus degraded voltage channels.

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 introduces no new modes of plant operation or new plant configuration that could lead to a new or different kind of accident from any previously evaluated being introduced. The 4160 V vital bus loss-of-voltage protection relays are required to operate following a complete loss of off-site power to initiate the bus power source transfer to on-site power, *i.e.,* the EDGs, to prevent a loss of all AC power. Likewise, the 480 V bus degraded voltage relays are required to operate upon detection of a sustained undervoltage condition to protect the Class IE components from damage from low voltage by initiating transfer of the 4160 V safety bus power source to the EDG. These safety functions are unchanged by the proposed setpoint revisions, and the proposed setpoints continue to provide the required actions consistent with the ANO–1 safety analysis.

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 two undervoltage relays located on each 4160 V safety bus are provided to detect loss-of-voltage, isolate the safety buses, initiate load shedding, and start the EDGs. The two undervoltage relays located on each 480 V safety bus are provided to detect sustained undervoltage, isolate the safety buses, and start the EDGs. These safety functions are unchanged by the proposed setpoint revisions. The proposed changes to the allowable values for both loss-of-voltage and degraded voltage relays incorporate channel uncertainties and calibration tolerances, while fully meeting their required safety functions of loss-of-voltage and degraded voltage protection without resulting in undesired tripping of the offsite power source.

The lower loss-of-voltage values do not affect the margin of safety since there is no appreciable time difference in reaching the lower setpoints during a loss-of-voltage event. The maximum proposed time delay allowable value with the minimum loss-of-voltage relay allowable value is within that used in the ANO–1 safety analysis. The revised allowable values for the loss-of-voltage relays will continue to provide the safety function with no appreciable additional time delay. Additionally, the lower voltage settings will help to prevent unnecessary isolation from the off-site power sources due to off-site perturbations in the electrical grid, and thus contribute to increasing the margin of safety. Also, the slightly higher range of allowable values for the degraded voltage settings allows enhanced protection of the Class IE components, but does not result in undesired tripping of the offsite power source for the analyzed grid minimum normal condition. The degraded voltage relays, therefore, also act to contribute to an increased margin of safety.

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

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, 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.

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: September 28, 2000.

Description of amendment request: The proposed changes to Arkansas Nuclear One, Unit 1 (ANO–1), Technical Specifications (TS) provide for the implementation of a revised reroll repair process for ANO–1 Once-Through Steam Generators (OTSG). The current TSs limit application of the reroll repair process to repair tubes with defects in the upper tubesheet area only, using a 1 inch roll length, and allow the reroll repair process to be performed only once per steam generator tube. The requested amendment would allow the reroll repair process to be used multiple times for a single tube and would allow the repairs in both the upper and lower tubesheets.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the criteria in 10 CFR 50.92(c).

OTSG tubesheet areas where reroll installation is excluded are specified in Appendix A of topical report BAW–2303P, Revision 4 [A non-proprietary version of the report, BAW–2303NP, Revision 4, ''OTSG Repair Roll Qualification Report,'' was submitted on October 26, 2000.]. The following discussion applies to areas of the OTSG tubesheets where installation of reroll repairs is permitted:

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

Two types of repair rolls have been developed for installation in the OTSGs, a single 1-inch roll expansion and an overlapping roll consisting of two 1-inch roll expansions. The overlapping roll provides a minimum of 1⅝ inch effective roll expansion. There is an additional ¼-inch roll transition region on each end of the roll expansion and a new leak-limiting pressure boundary is created by the repair roll. Applicable OTSG transient conditions were evaluated to develop a set of bounding test conditions for application to both types of repair rolls. Testing included examination of the effects of crevice deposits, cyclic loading, tube yield strength, differential dilations, axial loads and internal pressure.

Test results conclude that the single 1-inch minimum repair roll is structurally adequate to prevent tube slip during all non-faulted operating transients. A small amount of slippage is acceptable provided the tube does not slip out of the tubesheet and tube bow due to post-faulted transient heatup does not result in tube failure. Exclusion areas are established in the tubesheets to provide assurance that tube will not slip out of the tubesheet . The 1⅝ inch minimum overlapping roll is structurally acceptable based on the bounding evaluation of the single 1-inch repair roll.

Bounding leak rates are applied based on tubesheet depth and radial position. A post-slip leak rate is applied to any location where there is potential for repair roll slip during a postulated accident. The bounding leak rates are very conservative because the leakage is based on test samples with a full circumferential sever outboard of the repair roll. The majority of the degradation in the tubesheets is comprised of short, axial cracks for which the leakage would be much less under axial tensile loads than for the tested severed tube. In addition, repair rolls will actually slip only if the tube is severed outboard of the repair roll. Since the majority of the degradation in the region of the roll joints has been identified as small axial cracks, the probability of the repair roll maintaining structural integrity is very high and the potential for a joint to slip is very low. The leakage from each repair roll that serves as a pressure boundary is added to the leakage from all other sources and the total leakage must be within current accident analysis limits.

The application of the reroll repair process as described in topical report BAW–2303P, Revision 4 will not alter the conditions assumed in the current ANO–1 accident analysis for OTSG tube failures under postulated accident conditions. In addition, the condition of the OTSG tubes in this region are monitored during regular inspection intervals to assess for evidence of degradation. Any degradation noted will be addressed in the operational assessment and appropriate actions taken.

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 reroll process establishes a new pressure boundary for the associated tube in the tubesheet region inboard of the flaw. The new roll transition may eventually develop primary water stress corrosion cracking (PWSCC) and require additional repair. Industry experience with roll transition cracking has shown that PWSCC in roll transitions are normally short axial cracks, with extremely low leak rates. The standard MRPC eddy current inspection during the refueling outages have proven to be successful in detecting these defects.

In the unlikely event the rerolled tube failed and severed completely at the heel transition of the reroll region, the tube would retain engagement in the tubesheet bore, preventing any interaction with neighboring tubes. In this case, leakage is minimized and is well within the assumed leakage of the design basis tube rupture accident. In addition, the possibility of rupturing multiple steam generator tubes is unaffected. 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 repair roll is applicable to repairing axial, volumetric, or circumferential indications. Testing was conservatively performed with the assumption that the tube is severed at the heel transition (360 degree and 100% through-wall circumferential defect). The joint strength margin (actual load/limiting load) was calculated for each tubesheet depth and radial position for the cooldown transient to ensure margin against slip for non-faulted conditions. All locations showed a joint strength margin less than 0.65 with an acceptable margin less than 1.0.

A tube with degradation can be kept in service through the use of the reroll process. The new roll expanded interface created with the tubesheet satisfies all of the necessary structural and leakage requirements. Since the joint is constrained within the tubesheet bore, there is no additional risk associated with tube rupture. Therefore, the analyzed accident scenarios remain bounding, and the proposed modifications to the reroll process do not reduce the margin of safety.

Therefore, 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, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 9, 2000.

Description of amendment request: The proposed amendment requests fourteen of the simpler, generic administrative/editorial/consistency improvements agreed upon between the Nuclear Energy Institute Technical Specification Task Force (TSTF) and the NRC, subsequent to the conversion of the Perry Technical Specifications to the improved Standard Technical Specifications. The proposed amendment requests Perry-specific versions of TSTF 5, ''Delete Notification, Reporting, and Restart Requirements if a Safety Limit is Violated;'' TSTF 32, ''Slow/Stuck Control Rod Separation Criteria;'' TSTF 38, ''Revise Visual Surveillance of Batteries to Specify Inspection is for Performance Degradation;'' TSTF 52, ''Implement 10 CFR Part 50, Appendix J, Option B;'' TSTF 65, ''Use of Generic Titles for Utility Positions;'' TSTF 104, ''Relocate to the Bases the Discussion of Exceptions to Limiting Condition for Operation (LCO) 3.0.4;'' TSTF 106, ''Change to Diesel Fuel Oil Testing Program;'' TSTF 118, ''Administrative Controls Program Exceptions;'' TSTF 152, ''Revise Reporting Requirements to be Consistent with 10 CFR Part 20;'' TSTF 153, ''Clarify Exception Notes to be Consistent with the Requirement being Excepted;'' TSTF 166, ''Correct Inconsistency between LCO 3.0.6 and the Safety Functional Determination Program (SFDP) Regarding Performance of an Evaluation;'' TSTF 258, ''Changes to Section 5.0, Administrative Controls;'' TSTF 278, ''Battery Cell Parameters (LCO 3.8.6) includes more than Table 3.8.6–1 Limits;'' and TSTF 279, ''Remove the Words 'Including Applicable Supports' from the Description of the Inservice Testing Program.''

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

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

The proposed changes involve reformatting and rewording of the existing Technical Specifications to be consistent with regulations or other existing Technical Specifications, or the changes do not involve a change in intent. The proposed changes also involve Technical Specification requirements that are administrative rather than technical in nature. As such, this change does not affect initiators of previously evaluated events, or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This proposed amendment 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 new or eliminate old requirements on design or operation of the plant. The administrative changes also do not introduce new initiators of events. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change has no impact on any safety analysis assumptions or design basis margins. This change is administrative in nature. The proposed changes will not impose new or eliminate old requirements on design or operation of the plant. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

GPU Nuclear, Inc., Three Mile Island Nuclear Station, Unit 2, Docket No. 50–320, Dauphine County, Pennsylvania

Date of amendment request: July 25, 2000.

Description of amendment request: The proposed technical specifications change request (TSCR) is to revise Three Mile Island Nuclear Generating Station, Unit 2 (TMI–2), Technical Specification (TS) Section 6.7.2 to eliminate a change associated with periodic reviews of procedures. Currently, TS 6.7.2 states that required procedures shall be reviewed periodically as required by American National Standards Institute (ANSI) N18.7–1976 (a biennial review). This TSCR proposes to revise the wording for TS 6.7.2 to be essentially identical with the Three Mile Island, Unit 1 (TMI–1), TS requirements for procedure reviews, which states that required procedures shall be revised periodically, as set forth in administrative procedures (currently a biennial review). This TSCR would also be consistent with the TMI–2 Post-Defueling Monitored Storage (PDMS) Quality Assuance (QA) Plan, which states that ''Procedural documentation shall be periodically reviewed for adequacy as set forth in administrative procedures.''

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:

Applying the three standards set forth in 10 CFR 50.92, the proposed changes to the Technical Specifications involve no significant hazards consideration. The proposed changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators or assumptions are affected. The proposed changes have no effect on any plant systems. All Limited Conditions for PDMS and Safety Limits specified in the Technical Specifications will remain unchanged.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected. The proposed changes do not alter the source term, containment isolation, or allowable radiological consequences. The change in specified periodic procedure review requirements will have no adverse effect on any plant system.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. The proposed changes have no direct effect on any plant systems. The changes do not affect any system functional requirements, plant maintenance, or operability requirements.

3. Not involve a significant reduction in the margin of safety because the proposed changes do not involve significant changes to the initial conditions contributing to accident severity or consequences. The proposed changes have no direct effect on any plant systems.

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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 0037.

NRC Section Chief: Michael T. Masnik.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 16, 2000.

Description of amendment request: The proposed amendment would incorporate new pressure and temperature (P/T) curves into the Technical Specifications. The reactor pressure vessel P/T limit curves would be updated for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality.

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

(1) The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The P/T [pressure and temperature] limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the ASME [American Society of Mechanical Engineers] Code and 10 CFR 50 Appendix G and H as restrictions on operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary.

The changes to the calculational methodology for the P/T limits based upon Code Case N–640 continue to provide adequate margin in the prevention of a non-ductile type fracture of the reactor pressure vessel (RPV). The Code Case was developed based upon the knowledge gained through years of industry experience. P/T curves developed using the allowances of Code Case N–640 indeed yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects shows that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the Code Case in developing the P/T limit curves will continue to provide adequate protection against non-ductile type fractures of the RPV.

The proposed change will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events. The change does not adversely affect the integrity of the reactor coolant system such that its function in control of radiological consequences is affected.

(2) The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The amendment will revise the P/T curves which are established to the requirements of 10 CFR 50, Appendix G to assure that non-ductile fracture of the reactor vessel is prevented.

The proposed change provides more operating margin in the P/T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with benefits being primarily realized during the pressure tests. The proposed change does not result in any new or unanalyzed operation of any system or piece of equipment important to safety, and as a result, the possibility of a new type event is not created.

(3) The proposed amendment will not involve a significant reduction in a margin of safety. 10 CFR 50, Appendix G specifies fracture toughness requirements to provide adequate margins of safety during operation over the service lifetime. The values of adjusted reference temperature and upper shelf energy are expected to remain within the limits of Regulatory Guide 1.99, Revision 2 and Appendix G of 10 CFR 50 (less than 200 degrees F and greater than 50 ft-lbs respectively) for at least 32 effective full power years (EFPY) of operation.

The proposed change reflects an update of P/T curves based on the latest ASME guidance. The revised P/T curves provide more operating margin and thus, more operational flexibility than the current P/T curves. With the increased operational margin, a reduction in the safety margin results with respect to the existing curves. However, industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T curves are unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N–640 takes into account the acquired knowledge and establishes more realistic methodologies for the development of P/T curves. Therefore, operational flexibility is gained and an acceptable margin of safety to RPV non-ductile type fracture is maintained.

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

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

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: November 29, 1999, as supplemented on November 10, 2000.

Description of amendment request: The licensee submitted a proposed amendment to Kewaunee Nuclear Plant's Technical Specifications (TSs) modifying the TSs to incorporate requested changes per Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

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

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

The Shield Building Ventilation, the Auxiliary Building Ventilation, the Spent Fuel Pool Sweep Systems and the Control Room Post Accident Recirculation System are not accident initiators. Therefore, the proposed change will not increase the probability of an accident. The purpose of each of these systems is to mitigate the consequences of an accident once it has occurred. Based upon a comparison, the later version of ASTM D3803, ASTM D3803–89 was found to test the efficiency of the charcoal material under more conservative conditions. By testing the charcoal absorber material under more conservative conditions, the charcoal will require replenishment sooner. Therefore, the consequences will not be increased.

The changes to the basis sections are to promote clarity and uniformity. These statements were previously contained in the basis section or clarify which revision of Regulatory Guide 1.52 that should be used. This change provides acceptable guidelines for the qualification of replacement charcoal absorbent. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

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

This amendment request does not change any component at the plant. It is changing the testing requirements for material already installed. The material being tested has not changed. By testing the charcoal material under this revised protocol the material will be replaced with fresh charcoal sooner. This will ensure the equipment performs as described in the USAR.

The changes to the basis sections are to promote clarity and uniformity. These statements were previously contained in the basis section or clarify which revision of Regulatory Guide 1.52 that should be used. This change provides acceptable guidelines for the qualification of replacement charcoal adsorbent. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no reduction in the margin of safety. The efficiency of the charcoal material assumed by the USAR will not change as a result of this amendment and the functioning of the system will not change. Therefore, the original margin of safety is maintained.

The changes to the basis sections are to promote clarity and uniformity. These statements were previously contained in the basis section or clarify which revision of Regulatory Guide 1.52 that should be used. This change provides acceptable guidelines for the qualification of replacement charcoal adsorbent. Therefore, these changes will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: October 12, 2000.

Description of amendment request: In accordance with 10 CFR 50.59, the topical report WPSRSEM–NP, ''Reload Safety Evaluation Methods for Application to Kewaunee,'' Revision 3, is being submitted for the staff's review and approval since the licensee determined the revision of the report involved an unreviewed safety question. The topical report is intended to be applicable to Kewaunee reload cycles after and including Cycle 25, presently scheduled to commence in the fall of 2001. The topical report reflects:

- Editorial changes, including corrections to the limiting directions of core physics parameters and clarification of the definition of core physics parameters.
- Changes made to incorporate the CONTEMPT code for containment analysis. CONTEMPT is currently described for this purpose in the Kewaunee updated safety analysis report (USAR).
- The adoption of the GOTHIC code for containment analysis.
- Changes in Reload Safety Evaluation Methods due to Large Break Loss-of-Coolant Accident Upper Plenum Injection Analysis.
- The adoption of RETRAN–3D for use in the 2D mode for system analysis.
- The extension of the VIPRE–01 code to reflect changes in fuel design.

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

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

Analysis methods are not accident initiators, therefore, changes in analysis methods will not increase significantly the probability of occurrence or the consequences of an accident previously evaluated.

The changed analysis methods are conservative and conform to industry standards for analysis methods that are applied to design basis safety analyses. Benchmark analyses have demonstrated good agreement between the changed analysis methods and the current analysis of record (AOR) methods. The safety analysis results using the changed analysis methods are shown to satisfy all applicable design and safety analysis acceptance criteria. The demonstrated adherence to safety analysis acceptance criteria precludes new challenges to components and systems that could adversely affect the ability of existing components and systems to mitigate the consequences of any accident or adversely affect the integrity of any fission product barrier.

Analysis methods changes will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. The analysis methods changes also do not affect the plant configuration or the overall plant performance capabilities.

Therefore, the changes will not increase probability or the consequences of an accident previously evaluated.

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

The proposed change is a change to the analysis methods, which are applied to Kewaunee. Analysis methods are not accident initiators. The changed analysis methods are applied to the accidents that are the established design basis accidents for Kewaunee. Analysis methods changes will not impact plant equipment important to safety. Equipment important to safety will continue to operate within its design capabilities. The analysis methods changes also do not affect the plant configuration or the overall plant performance capabilities.

As demonstrated by the benchmark reports the methodologies provide a more accurate but still conservative representation of expected plant response following a design basis accident. Since the new methodologies are conservative with respect to actual expected plant response the changes will not create the possibility of an accident of a different type than any previously evaluated.

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

The proposed changes are changes to the analysis methods, which are applied to Kewaunee design basis safety analyses. The revised analysis methods have been verified through benchmark analyses against the current Analysis of Record methods. The analysis methods are conservative and appropriate for application to Kewaunee design basis analyses. Safety analysis acceptance criteria are satisfied when the changed analysis methods are applied to the Kewaunee design basis safety analyses. Demonstrated adherence to safety analysis acceptance criteria using the new analysis methods assures that Technical Specification limits will be satisfied during operation with the changed analysis methods.

Therefore, the margin of safety as defined in the basis of any Technical Specification will not be reduced significantly because of these changes.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: November 10, 2000.

Description of amendment request: The proposed amendment is to revise several sections of the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs). These sections include administrative changes, Table 4.1–1, and Sections 1.0, 6.4, and 6.10.

Administrative changes are submitted with this proposed amendment to correct minor typographical errors in the Table of Contents and among these changes are renumbering the index section pages and the addition of previously omitted sections.

The proposed changes will modify TS Table 4.1–1, ''Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels.'' This proposed change will decrease the calibration frequency for Turbine First Stage Pressure to support KNPP's 18-month operating cycle, and modify the table to eliminate a note that could lead to non-conservative calibration frequency.

The proposed TS Section 1.0, ''Definitions,'' will incorporate a line item improvement to provide additional clarification on channel calibration.

The proposed TS Section 6.4, ''Training,'' will remove the title of director for the KNPP training program and relocate the title reference to the Operational Quality Assurance Program Description (OQAPD).

The proposed TS Section 6.10, ''Record Retention,'' will revise the off-site review committee title.

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

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

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The proposed changes are administrative in nature and, therefore, have no impact on accident initiators or plant equipment, and thus do not affect the probability or consequences of an accident.

TS Section 1.0, ''Definitions''

A calibration will continue to ensure that a channel is within specification. Furthermore, calibration methodology is not an accident initiator. Therefore, the proposed change will not significantly raise the probability or consequences of an accident previously evaluated.

TS Table 4.1–1

The proposed change amends the calibration interval of the turbine first stage pressure from 12 months to each refueling cycle to coincide with KNPP's operating cycle. Calibration frequency would not change the consequence of a failure of the first stage pressure channel. Calibration frequency is not an accident initiator. Therefore, the proposed changes will not significantly raise the probability or consequences of an accident previously evaluated. Additionally, this change is consistent with the turbine first stage pressure calibration frequency stated in STS.

The proposed changes to the identified line items in Table 4.1–1 will require calibration of the instruments on a refueling cycle interval without exception. These calibration frequencies are not accident initiators and thus do not affect the probability of an accident. These changes are more conservative than existing TS and, therefore, will not increase the consequences of an accident.

TS Section 6.4, ''Training''

The proposed change will not change the intent of the TS. Removing the title from the TS is administrative in nature and, therefore, has no impact on accident initiators or plant equipment, and thus does not affect the probability or consequences of an accident.

TS Section 6.10, ''Record Retention''

The proposed change will not change the intent of the TS. Changing the title of the off-site review committee is administrative in nature and, therefore, has no impact on accident initiators or plant equipment, and thus does not affect the probability or consequences of an accident.

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

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The proposed changes do not involve changes to the physical plant or operations. Since these administrative changes do not contribute to accident initiation, they do not produce a new accident scenario or produce a new type of equipment malfunction. Also, these changes do not alter any existing accident scenarios; they do not affect equipment or its operation, and thus, do not create the possibility of a new or different kind of accident.

TS Section 1.0, ''Definitions''

The proposed TS change to channel calibration will not introduce any new equipment or result in existing equipment functioning differently from that previously evaluated in the USAR or TS. Calibration will continue to ensure that the channel is within specification and capable of performing its design basis function. No new accident is introduced and no safety-related equipment or safety functions are altered. Therefore, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accident.

TS Table 4.1–1

The proposed TS change will not introduce any new equipment or result in existing equipment functioning differently from that previously evaluated in the USAR or TS. The proposed change amends the calibration interval of the turbine first stage pressure from 12 months to each refueling cycle to coincide with KNPP's operating cycle. Performing the surveillance during refueling will decrease the likelihood for an induced transient. Expanding the calibration frequency will not affect the performance of the first stage pressure channel. A review of turbine first stage pressure calibration results for the last three years concluded no adjustment of the instrument was necessary due to little or no drift. Furthermore, similar transmitters already calibrated on a refueling basis have remained within acceptable limits. These results indicate stable instrument performance to support extending calibration frequency from 12 months to each refueling cycle.

The proposed changes will ensure that the affected channels are calibrated on a refueling basis. These changes will not introduce any new equipment or result in existing equipment functioning differently from that previously evaluated in the USAR or TS. No new accident is introduced and no safety-related equipment or safety functions are altered. The proposed changes do not affect any of the parameters or conditions that contribute to initiation of any accident.

TS Section 6.4, ''Training''

The proposed change does not involve a change to the physical plant or operations. Since an administrative change does not contribute to accident initiation, it does not produce a new accident scenario or produce a new type of equipment malfunction.

TS Section 6.10, ''Record Retention''

The proposed change does not involve a change to the physical plant or operations. Since an administrative change does not contribute to accident initiation, it does not produce a new accident scenario or produce a new type of equipment malfunction.

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

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Administrative changes do not involve a significant reduction in the margin of safety. The proposed changes do not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by these changes.

TS Section 1.0, ''Definitions''

Operation of the facility in accordance with this proposed TS change would not involve a significant reduction in a margin of safety. The specification will still ensure the operability of channels requiring calibration.

TS Table 4.1–1

Operation of the facility in accordance with the proposed TS changes would not involve a significant reduction in the margin of safety. The calibration will continue to verify the operability of the turbine first stage pressure channels. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Operation of the facility in accordance with the proposed TS changes would not involve a significant reduction in a margin of safety. The proposed changes will ensure the continued reliability of the instruments. This change is more conservative than existing TS and is consistent with STS.

TS Section 6.4, ''Training''

Administrative changes do not involve a significant reduction in the margin of safety. The proposed change does not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by this change.

TS Section 6.10, ''Record Retention''

Administrative changes do not involve a significant reduction in the margin of safety. The proposed change does not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by this change.

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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: March 20, 2000.

Description of amendment request: The proposed amendment would revise the Susquehanna Steam Electric Station (SSES), Unit 2, Technical Specification 2.1.1.2, minimum critical power ratio (MCPR) safety limits. These safety limits are being revised to reflect planned changes to the core composition for the next operating cycle and to support a separate license amendment proposing an increase in the SSES, Unit 1 and 2, 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes in MCPR Safety Limits do not affect any plant system or component (except the reactor core) and therefore does not increase the probability of an accident previously evaluated.

A Unit 2 Cycle 11 MCPR Safety Limit analysis was performed for PPL by SPC [Siemens Power Corporation]. This analysis used NRC approved methods as required by SSES Technical Specifications. For Unit 2 Cycle 11 [U2C11], the critical power performance of the ATRIUM™–10 fuel was determined using the NRC approved ANFB–10 correlation. Also, the analysis for U2C11 supports a Core Thermal Power of 3493 MWt which is a 1.5% increase over U2C10 (3441 MWt). The Safety Limit MCPR calculations statistically combine uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and uncertainties in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The resulting two-loop and single-loop MCPR Safety Limits are included in the proposed Technical Specification change. Thus, the cladding integrity and its ability to contain fission products are not adversely affected. It is therefore concluded that the proposed change does not increase the consequences of an accident previously evaluated.

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

As discussed above, the proposed changes to the Unit 2 Technical Specifications (MCPR Safety Limits) do not affect any plant system or component and do not affect plant operation. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since the proposed changes do not affect any plant system or component, and do not have any impact on plant operation, the proposed changes will not affect the function or operation of any plant system or component. The consequences of transients and accidents will remain within the criteria approved by the NRC. The proposed MCPR Safety Limits do not involve a significant reduction in the margin of safety as currently defined in the bases of the applicable Technical Specification sections. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Marsha Gamberoni.

Southern California Edison Company, et al., Docket No. 50–206, San Onofre Nuclear Generating Station, Unit 1, San Diego County, California

Date of amendment requests: October 30, 2000–PCN 268.

Description of amendment requests: This amendment application requests to delete license condition 2.C(3) related to fuel transshipments between San Onofre Nuclear Generating Station, Unit 1 (SONGS 1), which is in the process of decommissioning, and SONGS Units 2 or 3 since such transshipments will no longer be made. In addition, the amendment application requests revisions to the Unit 1 defueled Technical Specifications to (1) remove the spent fuel pool (SFP) temperature limits and related cooling system operability requirements, (2) remove the SFP auxiliary feedwater storage tank makeup water requirements and related surveillance requirements, (3) change the SFP water level limit for conditions other than spent fuel movement, and (4) change the operator staffing requirements for the decommissioning control room. As a result of these proposed changes, the licensee also proposes to delete the definitions of FUNCTIONAL and SPENT FUEL POOL COOLING TRAIN and revise the table of contents and list of tables according to the above changes.

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

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

No. The proposed change is a request to revise the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) license and permanently defueled technical specifications. The license condition for transshipment is being deleted since there is no safety-related equipment to protect and no plans for transshipment of SONGS 1 fuel to SONGS 2 or 3. Since the purpose of removing this license condition is that this activity will no longer be performed, there is no impact on accident probability or consequences. Deleting the technical specifications for spent fuel pool temperature and makeup are based on the current benign status of the spent fuel and spent fuel pool. The requirements and surveillances provided by these technical specifications no longer provide appropriate limits for the safe storage of the spent fuel. The spent fuel temperature limit cannot be reached. Makeup water is available from various sources onsite and offsite in a timely manner. Deleting these technical specifications has no impact on the probability or consequences of an accident. Modifying the spent fuel pool water level requirements provides two levels for maintaining water: One water level (elevation 28' [feet]) for just storage and a higher water level (elevation 40′ 3″ [inches]) for fuel movement. Lowering the water level for storage of spent fuel does not affect the accident probability. The fuel handling accident will not occur when the pool water level is at elevation 28 feet since spent fuel will only be handled when the pool water level is at elevation 40′ feet 3″. Removing the restrictions for having one individual of the minimum shift crew located in the control room will not have any impact on the fuel handling accident since a certified fuel handler is still required to be present.

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

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

No. This proposed change is a request to revise the SONGS 1 license and permanently defueled technical specifications. The transshipment license condition is being deleted since there is no safety-related equipment to protect and no plans for transshipment of Unit 1 spent fuel to Units 2 or 3. The technical specifications for spent fuel pool temperature and makeup are being deleted since these requirements no longer provide limits appropriate for maintaining the spent fuel pool. Removing these requirements does not create the possibility for a new or different accident since the associated limits are no longer attainable by the spent fuel pool. The only potential accident remaining is the spent fuel handling accident. Lowering the level of the spent fuel pool to elevation 28 feet has no impact on accident initiations since fuel handling will not be allowed at this water level. Removing the restrictions in the location of the minimum shift crew has no impact on accident initiation, and the certified fuel handler will be present during fuel handling operations.

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

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

No. This proposed change is a request to delete requirements from the license and the technical specifications and modify the spent fuel pool level requirements. Deleting the transshipment license condition has no impact on margin since there no longer is any safety-related equipment to protect and there are no plans for transshipment of Unit 1 spent fuel to Units 2 or 3. Deleting the spent fuel pool temperature and makeup requirements has no effect on margin since the status of the spent fuel pool is such that the margins associated with these requirements have increased and with time will continue to increase. Modifying the level requirement to allow the water level to be at elevation 28 feet for spent fuel storage has no impact on margin since the spent fuel has cooled significantly and fuel movement will not occur at this level. Since the status of the spent fuel pool is such that the margins are improving with time, removing the restrictions in the location of the minimum shift crew has no effect on the margins.

Therefore, this change 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. The staff also reviewed the proposed administrative changes to delete definitions and conform the table of contents and list of tables to the proposed changes for no significant hazards consideration. These administrative changes do not affect the design or operation of the facility and satisfy the three standards of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Michael Masnik (Unit 1).

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

Date of application for amendments: October 5, 2000.

Brief description of amendments: The amendments revise the VEGP Updated Final Safety Analysis Report (UFSAR) Chapters 11 and 15 to incorporate changes due to an updated Dose Equivalent Iodine analysis. The new analysis was performed in response to Westinghouse Nuclear Safety Advisory Letter, ''NSAL–00–04: Nonconservatisms in Iodine Spiking Calculations.''

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

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the [Updated Final Safety Analysis Report] UFSAR. The comprehensive engineering review included evaluations or reanalysis of all accident analyses. The letdown flow rate does not initiate any accident; therefore, the probability of an accident has not been increased. All dose consequences have been analyzed or evaluated with respect to the proposed changes, and all acceptance criteria continue to be met. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the UFSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, all accident analyses criteria continue to be met, and these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety. All analyses and evaluations using these inputs have been revised to reflect the proposed values. The evaluations and analyses results demonstrate that applicable acceptance criteria are met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: November 3, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.11, ''Technical Specification Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 which were published in the **Federal Register** (64 FR 53582) on October 4, 1999.

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

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

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revision of 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the critieria in paragraph 10 CFR 50.59(c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: November 6, 2000.

Brief description of amendments: The request revises the VEGP Technical Specification (TS) Limiting Conditions for Operation 3.7.10, 3.7.11, and 3.7.13 to address degraded pressure boundaries. The changes revise the TS to allow the pressure boundaries of ventilation systems such as the Control Room Emergency Filtration System (CREFS) and the Piping Penetration Area Filtration and Exhaust System (PPAFES) to be opened intermittently under administrative controls. A new condition is also added that allows 24 hours to restore inoperable CREFS and PPAFES pressure boundaries before requiring the units to perform an orderly shutdown.

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?

No. The control room emergency filtration system (CREFS) and the piping penetration area filtration and exhaust system (PPAFES) are not assumed to be initiators of any analyzed accident. Therefore, the proposed changes do not affect the probability of any accident previously evaluated. The proposed changes for the CREFS and PPAFES Technical Specifications (TS) would permit the subject pressure boundaries to be opened intermittently under administrative control. Based on the proposed compensatory measures in the form of a dedicated individual who is in communication with the control room, and his ability to rapidly restore the pressure boundary, the capability to mitigate a design basis event will be maintained. In addition, the proposed changes would add a new condition that would permit a 24-hour period to take action to restore an inoperable pressure boundary to operable status, modify existing conditions to accommodate the new condition (so as to maintain the requirements of the existing conditions), and correct a typographical error. With respect to CREFS, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated based on the availability of a self-contained breathing apparatus to minimize radiological dose due to iodine and the ability to operate more than one train as the need arises to maintain positive pressure or at least maintain an outflow of air from the control room environment. With respect to the PPAFES, it has been demonstrated by analysis that a breach of the pressure boundary will not result in control room or offsite doses that exceed their respective limits. The correction of the typographical error is an administrative change that has no technical impact.

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

No. The proposed changes for the CREFS and PPAFES TS would permit the subject pressure boundaries to be opened intermittently under administrative control. In addition, the proposed changes would add a new condition that would permit a 24-hour period to take action to restore an inoperable pressure boundary to operable status, modify existing conditions to accommodate the new condition (so as to maintain the requirements of the existing conditions), and correct a typographical error. The proposed changes do not alter the operation of the plant or any of its equipment, introduce any new equipment, or result in any new failure mechanisms or single failures. Therefore, there is no potential for a new accident and no changes to the way that an analyzed accident will progress. The correction of the typographical error is an administrative change that has no technical impact.

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

No. The proposed changes for the CREFS and PPAFES TS would permit the subject pressure boundaries to be opened intermittently under administrative control. In addition, the proposed changes would add a new condition that would permit a 24-hour period to take action to restore an inoperable pressure boundary to operable status, modify existing conditions to accommodate the new condition (so as to maintain the requirements of the existing conditions), and correct a typographical error. The proposed changes do not adversely affect the ability of the fission product barriers to perform their functions. The only safety-related equipment affected by the proposed changes is the CREFS and the PPAFES. It has been demonstrated by analysis that a breach in the pressure boundary of the PPAFES will not cause the control room or offsite doses to exceed their respective limits. Adequate compensatory measures are available to mitigate a breach in the CREFS pressure boundary. The probabilities of design bases accidents that would place demands on these systems during a period that the ventilation system pressure boundaries would be allowed to be inoperable have been shown to be negligible. In addition, the proposed changes avoid the potential of placing one or both units in TS Limiting Condition for Operation (LCO) 3.0.3 solely due to a breach of the ventilation system pressure boundary. The correction of the typographical error is an administrative change that has no technical impact.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: November 16, 2000.

Brief description of amendments: The request proposes to amend Technical Specification 5.5.1, '' Technical Specification Bases Control Program'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999.

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

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

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revision of 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph 10 CFR 50.59 (c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of amendment request: November 21, 2000 (TSC–396).

Description of amendment request: The proposed amendment would revise the reactor core Safety Limit Minimum Critical Power Ratio (SLMCPR) specified in Technical Specification (TS) Section 2.1.1.2 from 1.10 to 1.07 for two reactor recirculation loop operation and from 1.12 to 1.10 for single loop operation. The change is based on use of newly approved analytical methodology for the Cycle 12 reload analysis. This methodology is described in Global Nuclear Fuels (GNF) licensing document, ''General Electric Standard Application for Reactor Fuel, GESTAR-II, Amendment 25,'' dated June 2000, which has been approved by NRC.

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 proposed amendment establishes revised SLMCPR values for two recirculation loop operation and for single recirculation loop operation. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Since the change does not require any physical plant modifications or physically affect any plant components, no individual precursors of an accident are affected and the probability of an evaluated accident is not increased by revising the SLMCPR values.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The revised SLMCPRs have been performed using NRC-approved methods and procedures. The basis of the MCPR [minimum critical power ratio] Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. These calculations do not change the method of operating the plant and have no effect on the consequences of an evaluated accident. Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed license amendment involves a revision of the SLMCPR for two recirculation loop operation and for single loop operation based on the results of an analysis of the Cycle 12 core. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in the allowable methods of operating the facility. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable methods of operation. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in the TS bases will remain the same. The new SLMCPRs are calculated using NRC-approved methods and procedures which are in accordance with the current fuel design and licensing criteria. The SLMCPRs remain high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: October 25, 2000.

Description of amendment request: The proposed change would revise the 125 volt DC (Vdc) station battery system Technical Specifications (TSs) to reflect the availability of a second, fully qualified charger, for each main station battery system. The licensee also proposed corresponding changes to the listing of components in the TSs.

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

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

There is no change in the method of operation of the 125 Vdc main station battery systems by this change. The battery chargers will function the same, except that an additional battery charger will be available to each system. No change to accident assumptions or precursors are involved with this change. Likewise, no change in system operation or response to analyzed events is affected.

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

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

The new chargers to be installed will provide additional charging capability. No reduction in DC system equipment operation or capability is involved. The methods by which the DC systems perform their safety functions are unchanged and remain consistent with current safety analysis assumptions. There is no change in system or plant operation that involves failure modes other than those previously evaluated.

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

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

No adverse affect on equipment operation or capability will result from this change. The installation of additional chargers in fact enhances the reliability of the battery charging function. The equipment fed by the DC systems involved in this change will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 1, 2000.

Description of amendment request: The proposed change would revise the operability requirement for high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) low steam line pressure isolation instrumentation to coincide with system operability requirements. The proposed change eliminates the need to open manual containment isolation valves under administrative control during reactor heatup, reduces the potential for operator error when closing these valves (potential for leaving valve mispositioned) and clarifies the steam line low pressure isolation function description. An administrative change to correct the HPCI High Steam Line d/p instrument component numbers was also proposed to ensure the accuracy of isolation instrumentation information.

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

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

The proposed change clarifies the equipment protection purpose of the HPCI and RCIC low steam line isolation function. [The proposed change would require] operability of the steam supply pressure instrumentation [ ] whenever the systems are required to be operable. This change does not significantly alter the function of containment isolation actuation instruments nor does it significantly alter containment integrity requirements. The proposed change does not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is being introduced.

The proposed change does not affect the ability of the primary containment isolation system or high pressure core cooling systems to perform their safety functions. The essential safety function of providing primary containment integrity is maintained since operability of the primary instrumentation associated with detection of a HPCI or RCIC steam line break outside containment will continue to be required when primary containment integrity is required. The essential safety function of providing water to cool the core in the event of a small break in the nuclear system is maintained. The operational change being made would not alter the sequence of events, plant response, or conclusions of existing safety analyses. This proposed change results in no impact on analyzed accident event precursors, initiators or effects.

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

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

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. No new or different types of equipment will be installed. Operation with the HPCI and RCIC steam line isolation valves open between 212 °F and 150 psig does not alter the input or result of existing accident analyses. The change in plant operation does not involve failure modes other than those previously evaluated. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

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

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

The change involves operation with the HPCI and RCIC systems with steam line isolation valves open between 212 °F and 150 psig. This change will not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed change maintains design margins of the primary containment isolation system or high pressure core cooling systems to perform their required safety functions. The essential safety functions of providing primary containment integrity and providing water to cool the core in the event of a small break in the nuclear system are maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or margins in existing safety analyses. This proposed change results in no impact on analyzed accident event precursors or effects.

This proposed change does not alter the physical design of the plant. The change in method of operation results in no significant impact on safety functions or assumed responses. The proposed change does not alter the means by which primary containment isolation is maintained and high pressure core cooling systems are operated.

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

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

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

NRC Section Chief: James W. Clifford.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: September 27, 2000, as supplemented November 21, 2000.

Description of amendment request: The proposed changes will increase the fuel enrichment limit from 4.3 weight percent to 4.6 weight percent Uranium^{235} (U^{235}), establish Technical Specifications to control the boron concentration in the spent fuel pool (SFP) and impose restrictions on the storage locations for some spent fuel assemblies, and change the method of criticality calculation used to evaluate the effect of a fuel enrichment change on the SFP.

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.] Criterion 1. The proposed increase in maximum fuel enrichment and the changes to the SFP design basis will not significantly increase the probability of or consequences of an accident previously evaluated in the North Anna Units 1 and 2 UFSAR [Updated Final Safety Analysis Report].

The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment and SFP changes involve criticality events during fuel handling and storage (e.g., fuel mispositioning). The proposed Technical Specifications establish additional restrictions on the placement of each fuel assembly in the SFP to ensure subcriticality. However, criticality safety analyses have been performed that demonstrate that the K_{eff} during the handling and storage of both new and spent fuel remains low enough to ensure subcriticality during postulated accident conditions. In addition, analyses of the dilution of the spent fuel pool have been performed to ensure that there is adequate time for a dilution event to be detected and mitigated, such that the required subcritical margin is maintained in the spent fuel pool. Therefore the probability of occurrence of criticality during fuel handling or storage is not significantly increased. In addition the consequences of the operating reactor accident scenarios are also unchanged, because the source terms used to determine the releases from fuel during accidents are a function of burnup, rather than initial enrichment.

[2.] Criterion 2. The proposed increase in maximum fuel enrichment or the change in the SFP design basis does not create a new or different kind of accident from any already discussed in the North Anna Units 1 and 2 UFSAR.

Although there are new restrictions on placement of fuel in the SFP, the administrative controls on fuel movement to specified locations in the pool are unchanged. The higher enrichment fuel and the new Technical Specifications for the spent fuel pool do not require any new or different plant equipment, and do not change the manner in which currently installed equipment is operated. There are no changes to normal core operation, and the units will meet all applicable design criteria and will operate within existing Technical Specifications limits. No new failure modes have been created for any system, component, or piece of equipment, and no new single failure mechanisms have been introduced. No new or different plant equipment is introduced, and the operation of currently equipment is not changed. The use of a higher maximum fuel enrichment will not cause the design criteria for fuel operation or storage to be exceeded. No new modes or limiting single failures are created by the use of a higher fuel enrichment. Safety analyses for the fuel storage area have demonstrated that subcriticality will be maintained during fuel handling and storage, including fuel mispositioning and pool dilution scenarios.

[3.] Criterion 3. The proposed increase in maximum fuel enrichment and the changes to the SFP design basis will not significantly reduce the margin of safety.

The use of higher enriched fuel and the changes to the SFP design basis have the potential to affect only criticality events during fuel handling and storage. Criticality analyses demonstrate that the limits on K_{eff} for the new and spent fuel storage areas will be satisfied. Therefore, there is adequate margin to ensure subcriticality during the storage and handling of fuel. The requirements of 10 CFR 50 Appendix A General Design Criterion 62 are satisfied. Safety analyses demonstrated that K_{eff} will remain sufficiently low to ensure subcriticality, so no new releases will result and there is no impact on radiological consequences of accidents. The safety analyses of record will remain applicable for the operation of fuel with a higher initial U^{235} enrichment and changes to the spent fuel pool. Therefore, the margin of safety is not affected by the proposed increase in initial fuel enrichment or changes to the spent fuel pool design basis.

Based on the evaluations and analyses results presented in the foregoing safety significance evaluation, it has been demonstrated that increasing the North Anna Units 1 and 2 maximum initial fuel enrichment to 4.6 weight percent U^{235} and changing the design basis of the spent fuel pool to eliminate any credit for Boraflex but take credit for soluble boron in the pool will not result in 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard L. Emch, Jr.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly 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).

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: July 27, 2000.

Brief Description of amendments: The amendments change the Technical Specifications to allow one of each unit's Direct Current power subsystems to be inoperable when in Modes 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment.

Date of issuance: November 29, 2000.

Effective date: November 29, 2000.

Amendment Nos.: 211 and 238.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register**: September 20, 2000 (65 FR 56948). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 29, 2000.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: August 1, 2000.

Brief description of amendments: The amendments revise TS Section 3.7.15 and associated Bases, and Section 4.0 for the McGuire Nuclear Stations, Units 1 and 2, to allow the use of credit for soluble boron in spent fuel pool criticality analyses. The request is based on the NRC-approved Westinghouse Owners Group Topical Report WCAP–14416–NP–A, which provides generic methodology for crediting soluble boron.

Date of issuance: November 27, 2000.

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

Amendment Nos.: 197 and 178.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 18, 2000 (65 FR 62385). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 2000.

No significant hazards consideration comments received: No.

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

Date of application of amendments: October 18, 2000.

Brief description of amendments: The amendments revise the implementation date of Amendment Nos. 312, 312, and 312 from November 30, 2000, so that implementation will be on or before implementation of amendments resulting from the application that must be submitted by April 5, 2001. This submittal will be based on an engineering study that is being conducted to evaluate both the appropriate Keowee Hydro Unit out-of-tolerance surveillance criteria and resolve overshoot concerns.

Date of Issuance: November 27, 2000.

Effective date: As of the date of issuance.

Amendment Nos.: 317/317/317.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Implementation Date.

Date of initial notice in **Federal Register**: October 25, 2000 (65 FR 63896). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 25, 2000.

Brief description of amendment: The amendment changed the action statements for Technical Specification (TS) 3.8.2.2, A.C. Distribution—Shutdown, and TS 3.8.2.4, DC Distribution—Shutdown, by replacing the requirement to establish containment integrity within eight hours with a requirement to immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions. Related changes to the associated Bases were also made.

Date of issuance: November 28, 2000.

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

Amendment No.: 227.

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

Date of initial notice in **Federal Register**: July 12, 2000 (65 FR 43045). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 31, 2000.

Brief description of amendment: The Technical Specifications (TS) were revised by adding an additional Condition to ITS 3.3.11, Emergency Feedwater Initiation and Control System Instrumentation, regarding the required action to be taken for one or more Emergency Feedwater Initiation and Control System channels when up to two Reactor Coolant Pump status signals are inoperable.

Date of issuance: November 21, 2000

Effective date: November 21, 2000

Amendment No.: 194.

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

Date of initial notice in **Federal Register**: July 12, 2000 (65 FR 43047). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2000.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: June 8, 2000.

Brief description of amendments: The amendments allow the use of probabilistic risk assessment (PRA) techniques in evaluating the need for tornado-generated missile barriers; this provides an alternative to installing physical missile protection for those structures, systems, and components that are not physically protected from tornado-generated missiles.

Date of issuance: November 17, 2000.

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

Amendment Nos.: 247 and 228.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments approved revision of the UFSAR.

Date of initial notice in **Federal Register**: July 12, 2000 (65 FR 43049). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 2000.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 1, 2000, as supplemented October 27, 2000.

Brief description of amendments: The licensee proposed the following three changes:

(1) A one-time change to Unit 1 Technical Specification (TS) Surveillance Requirement (SR) 4.6.1.2 to add the following: ''A one-time exception to the requirement to perform post-modification Type A testing is allowed for the steam generators and associated piping, as components of the containment barrier. For this case, American Society of Mechnical Engineers (ASME) Section XI leak testing will be used to verify leak tightness of the repaired or modified portions of the containment barrier. Entry into MODES 3 and 4 following the extended outage that commenced in 1997, may be made to perform this testing.''

(2) A change to Unit 1 and Unit 2 TS SR 4.6.1.2 to add the phrase ''except as modified by NRC-approved exemptions'' to the requirement to perform testing in accordance with 10 CFR Part 50, Appendix J, Option B, and the September 1995 version of Regulatory Guide 1.163.

(3) A change to the Unit 1 and Unit 2 Bases TS SR 4.6.1.2 to add the phrase ''Regulatory Guide 1.163, dated September 1995, and Nuclear Energy Institute (NEI) document NEI 94–01, except as modified'' after the surveillance testing for measuring leakage rates are consistent with the Appendix ''J'' of 10 CFR Part 50.

Date of issuance: November 17, 2000.

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

Amendment Nos.: 248 and 229.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 20, 2000 (65 FR 56953). The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 2000.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: April 6, 2000, as supplemented November 13, 2000.

Brief description of amendments: The amendments would approve changes involving unreviewed safety questions to the Updated Final Safety Analysis Report to incorporate new methodology to be used in the analysis of high-energy line breaks at D. C. Cook.

Date of issuance: November 21, 2000.

Effective date: As of the date of issuance.

Amendment Nos.: 249 and 230.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 23, 2000 (65 FR 51355). The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 21, 2000.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 18, 2000, as supplemented November 10, 2000.

Brief description of amendments: The amendments revise Technical Specifications (TSs) 3/4.7.1.2, ''Auxiliary Feedwater [AFW] System,'' to change the description in the TSs surveillance requirement (SR) 4.7.1.2.d of the position for each automatic valve in the AFW system from the ''fully open'' position to the ''correct'' position.

Date of issuance: November 30, 2000.

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

Amendment Nos.: 250 and 231.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 25, 2000 (65 FR 63899) The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 20, 2000, as supplemented on September 25, 2000.

Description of amendment request: The amendment revises the Technical Specifications (TS) by removing the prescriptive requirement for determining the reactor coolant system flow rate by precision heat balance in Surveillance Requirement 4.2.5.3. The amendment also revises TS Table 2.2–1 to reflect the allowed calibration tolerance of the protection racks and noting that the Trip Setpoint for Functional Unit 12, Reactor Coolant Flow-Low reactor trip is based on an indicated value rather than a measured value.

Date of issuance: October 26, 2000.

Effective date: As of its date of issuance, and shall be implemented at commencement of Cycle 8 operation (scheduled for November 2000).

Amendment No.: 77.

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

Date of initial notice in **Federal Register:** August 9, 2000 (65 FR 48753) The supplemental letter provided clarifying information within the scope of the original application 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 October 26, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of application for amendments: June 6, 2000.

Brief description of amendment: The amendment deletes or modifies license conditions and confirmatory orders to reflect the permanently defueled condition of the unit.

Date of Issuance: November 15, 2000.

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

Amendment No.: 108.

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

Date of initial notice in **Federal Register:** July 26, 2000 (65 FR 46010). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 2000.

No significant hazards consideration comments received: No.

Nuclear Management Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: May 12, 2000.

Brief description of amendment: The amendment revises the Technical Specification 4.6.E.1.d safety/relief valve bellows monitoring system test frequency from quarterly to once per operating cycle.

Date of issuance: November 30, 2000.

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

Amendment No.: 114.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 28, 2000 (65 FR 39959). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 2000.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, 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: December 21, 1999, as supplemented May 2, 2000

Brief description of amendments: These amendments incorporate changes to the Technical Specifications (TSs) to more clearly define the requirements for service water (SW) system operability in accordance with the system configuration assumed in the SW system analysis. The application dated December 21, 1999, as supplemented May 2, 2000, superceded an application dated July 30, 1998, in its entirety. The December 21, 1999, application was submitted because the licensee performed additional analyses of the SW system subsequent to the submittal of the July 30, 1998, application, which necessitated additional changes to the TSs.

Date of issuance: November 17, 2000.

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

Amendment Nos.: 199 and 204.

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

Date of initial notice in **Federal Register**: February 23, 2000 (65 FR 9014). The May 2, 2000, supplemental letter provided additional clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 2000.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: June 14, 2000, as supplemented on October 12, 2000.

Brief description of amendments: The amendments modify the Salem Unit Nos. 1 and 2 Technical Specifications (TS), and allow PSEG Nuclear to use the Best Estimate Analyzer For Core Operations—Nuclear (BEACON) system at Salem to fulfill certain TS surveillance requirements that involve core power distribution measurements. BEACON is a core power distribution monitoring and support system based on a three dimensional nodal code. The system is used to provide data reduction for incore neutron flux maps, core parameter analysis and follow, and core prediction.

Date of issuance: November 6, 2000.

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

Amendment Nos.: 237 and 218.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 26, 2000 (65 FR 46014). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 24, 1999, as supplemented September 14, 2000.

Brief description of amendment: The amendment revises the Technical Specifications to implement Filtration, Recirculation, and Ventilation System and Control Room Emergency Filtration System charcoal filter testing requirements that are consistent with the U.S. Nuclear Regulatory Commission guidance delineated in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: November 17, 2000

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

Amendment No.: 130.

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

Date of initial notice in **Federal Register**: December 29, 1999 (64 FR 73096). The September 14, 2000, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original application. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 20, 2000 (PCN–488, supplement 1; supersedes application dated August 11, 1999).

Brief description of amendments: The amendments revised Technical Specifications surveillance requirements (SRs) related to the acceptance criteria for TS 3.3.7, ''Diesel Generator (DG)—Undervoltage Start,'' SR 3.3.7.3, which verifies operability of the loss of voltage and degraded voltage actuation circuits. The amendments replaced the analytical limits currently specified as acceptance criteria with allowable values, and deleted SR 3.3.7.4 on the basis that it is redundant with SR 3.3.7.3.

Date of issuance: November 29, 2000.

Effective date: November 29, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2–174; Unit 3–165.

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

Date of initial notice in **Federal Register**: August 23, 2000 (65 FR 51362). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 29, 2000.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: August 28, 2000.

Description of amendment request: The amendments revise the Units 1, 2 and 3 Technical Specifications (TS) to incorporate TS Task Force (TSTF) Items Nos. TSTF–71, TSTF–208, TSTF–222, TSTF–284, TSTF–258 and TSTF–364. TSTFs are changes to the Improved Standard TS that were initiated by the nuclear power industry and submitted to the NRC staff. A description of each of the six TSTFs proposed for implementation at Browns Ferry follows: (1) TSTF–71, Revision 2, adds an example of the application of the Safety Function Determination Program to the Bases for Limiting Conditions for Operation (LCO) 3.0.6. (2) TSTF–208, Revision 0, extends the allowed time to reach MODE 2 in LCO 3.0.3 from 7 hours to 10 hours. The change is based on plant experience regarding the time needed to perform a controlled shutdown in an orderly manner. (3) TSTF–222, Revision 1, clarifies Improved Technical Specification (ITS) Section 3.1.4, Control Rod Scram Times, Surveillance Requirements (SRs) to better delineate the requirements for testing control rods following refueling outages and for control rods requiring testing due to work activities. (4) TSTF–258, Revision 4, revises TS Section 5.0, Administrative Controls, to delete specific TS staffing requirement provisions for Reactor Operators (ROs), eliminates TS details for working hour limits, clarifies requirements for the Shift Technical Advisor position, adds regulatory definitions for Senior ROs and ROs, revises the Radioactive Effluent Controls Program to be consistent with the intent of 10 CFR Part 20, deletes periodic reporting requirements for mainsteam relief valve openings, and revises radiological area control requirements for radiation areas to be consistant with those specified in 10 CFR 20.1601(c). (5) TSTF–284, Revision 3, modifies Improved TS Section 1.4, Frequency, to clarify the usage of the terms ''met'' and ''performed'' to facilitate the application of SR Notes. Two new SR Examples, 1.4–5 and 1.4–6, are added to illustrate the application of the terms. (6) TSTF–364, Revision 0, revises Section 5.5.10, TS Bases Control Program, to reference 10 CFR 50.59 rather than ''unreviewed safety question.'' Also, editorial change WOG–ED–24, which substitutes ''require'' for ''involve'' in 5.5.10.b is made for consistency in usage.

Date of issuance: November 21, 2000.

Effective date: November 21, 2000.

Amendment Nos.: 239, 266, and 226.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68: Amendments revised the licenses.

Date of initial notice in **Federal Register**: October 4, 2000 (65 FR 59224)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 15, 2000.

Brief description of amendments: The proposed change replaces the general references currently provided in Technical Specification 5.6.6 for determining the reactor coolant system pressure and temperature limits with the requirement that the Pressure/Temperature Limits and Low Temperature Overpressure Protection System Setpoints shall not be revised without prior U.S. Nuclear Regulatory Commission approval.

Date of issuance: November 27, 2000.

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

Amendment Nos.: 81 & 81.

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

Date of initial notice in **Federal Register**: November 1, 2000 (65 FR 65351). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 19, 2000.

Brief description of amendment: The amendment revises the Technical Specifications to establish operability requirements to ensure that adequate reactor coolant inventory and sufficient heat removal capability exist during cold shutdown and refueling conditions.

Date of Issuance: November 17, 2000.

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

Amendment No.: 195.

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

Date of initial notice in **Federal Register**: October 18, 2000 (65 FR 62393). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 17, 2000.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: November 29, 1999, as supplemented August 31, 2000.

Brief description of amendments: The amendments revise the testing requirements in Technical Specification (TS) 4.7.7.1 and TS 4.7.8.1 to incorporate the American Society for Testing and Materials D3803–1989 standard and the application of a safety factor of 2.0 for the charcoal filter efficiency assumed in Virginia Electric and Power Company's design-basis dose analysis.

Date of issuance: November 20, 2000.

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

Amendment Nos.: 224 and 205.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 9, 2000 (65 FR 6413). The August 31, 2000, supplement provided clarifying information only, and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 20, 2000.

No significant hazards consideration comments received: No.

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

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–31541 Filed 12–12–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a draft of a new guide in its Regulatory Guide Series, with its related Standard Review Plan section. The Regulatory Guide Series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified by its task number, DG–1096 (which should be mentioned in all correspondence concerning this draft guide), is titled ''Transient and Accident Analysis Methods.'' This guide is being developed to describe a process that is acceptable to the NRC staff for the development and assessment of evaluation models that may be used to analyze transient and accident behavior.

Draft Standard Review Plan Section 15.0.2, ''Review of Analytical Computer Codes,'' is being developed to describe the review process for NRC staff and acceptance criteria for analytical models and computer codes used by licensees to analyze accident and transient behavior. This draft Standard Review Plan (SRP) section is intended to become Section 15.0.2 of NUREG–0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.''

This draft guide and draft standard review plan section have not received complete staff approval and do not represent an official NRC staff position.

Comments on both documents may be accompanied by relevant information or supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415–4737 or (800) 397–4209; fax (301) 415–3548; email <*PDR@NRC.GOV*>. Comments will be most helpful if received by February 15, 2001.

You may also provide comments or access these documents via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415–5905; e-mail *CAG@NRC.GOV.* Electronic copies of this draft guide, under Accession Number ML003770849, are available in NRC's Public Electronic Reading Room, which can also be accessed through NRC's web site, *WWW.NRC.GOV.* For information about the draft guide and the related documents, contact Mr. N. Lauben at (301) 415–6762; e-mail *GNL1@NRC.GOV.* For information about the draft standard review plan section, contact Mr. J.L. Staudenmeier at (301) 415–2869, email *JLS4@NRC.GOV*; or Mr. M.A. Shuaibi at (301) 415–2859, email *MAS4@NRC.GOV.*

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Regulatory guides and standard review plan sections are available for inspection at the Commission's Public Document Room, 11555 Rockville Pike, Rockville, MD. Requests for single copies of draft or final guides or SRP sections (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section; or by fax to (301) 415–2289, or by email to *DISTRIBUTION@NRC.GOV.* Telephone requests cannot be accommodated. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 28th day of November 2000.

For the Nuclear Regulatory Commission.

Farouk Eltawila,

Acting Director, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research.

[FR Doc. 00–31736 Filed 12–12–00; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

Retirement Plan for Manually Set Postage Meters

AGENCY: Postal Service.

ACTION: Notice of final plan.

SUMMARY: This notice of the final plan for the retirement of manually set postage meters clarifies the second phase of the plan to take postage metering to a higher level of security. The Postal Service recently completed the first phase of an overall Postal Service plan with the decertification of mechanical postage meters. Upon completion of the four phases of this plan, all meters in service will offer enhanced levels of security, thereby greatly reducing the Postal Service's exposure to meter fraud, misuse, and loss of revenue.

DATES: May 1, 2000.

FOR FURTHER INFORMATION CONTACT: Nicholas S. Stankosky, 703–292–3703.

SUPPLEMENTARY INFORMATION: In 1995 the Postal Service, in cooperation with all authorized postage meter manufacturers, began a phase-out, or decertification, of all mechanical postage meters because of identified cases of indiscernible tampering and misuse. Postal revenues were proven to be at serious risk. The completion of this effort, which resulted in the withdrawal of 776,000 mechanical meters from service, completed Phase I of the proposed plan for secure postage meter technology. Recent advances in postage meter technology offer high