at: *http://www.nrc.gov/SECY/smj/schedule.htm*.

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

Dated: November 9, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–29354 Filed 11–13–00; 2:18 pm]

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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 October 23, 2000, through November 3, 2000. The last biweekly notice was published on November 1, 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 December 15, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, 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).

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

Date of amendment request: November 22, 1999, as supplemented on September 11, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification Sections 4.5.D, "Containment Air Filtration System (CAFS)," 4.5.E, "Control Room Air Filtration System (CRAFS)," 4.5.F, "Fuel Storage Building Air Filtration System (FSBAFS)," and 4.5.G, "Post-accident Containment Venting System (PACVS)," to address the testing requirements in Generic Letter 99–02, "Laboratory Testing of Nuclear-Grade Activated Charcoal."

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

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

No. The proposed change would revise Section 4.5 to incorporate current NRC [Nuclear Regulatory Commission] testing requirements which affect how the charcoal would be tested in the laboratory. These changes would not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications would remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

No. The proposed changes would implement testing methodology for ventilation system charcoal in accordance with Generic Letter 99–02, but would not alter equipment performance criteria or standards. The safety analysis of the facility would remain complete and accurate, and would not be affected by the new charcoal testing requirements. There would be no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated would still be valid. The operating procedures and emergency procedures would be unaffected. Consequently no new failure modes would be introduced as a result of the proposed change. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. Since there would be no changes to the operation of the facility, to its physical design, or to the performance characteristics of any safety-related equipment, neither the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, nor Technical Specification bases would be affected. 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: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: September 5, 2000.

Description of amendment request: The amendment revises Technical Specification 3.3.5.1, 3.3.6.1 and 3.3.6.2. The proposed changes would add notes to tables listing instrument channels that are common to, or support the operability of interrelated systems as governed by these technical specifications.

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

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

The proposed change has no impact on previously analyzed accidents or transients and has no affect on design, operation, capacity, or surveillance requirements of the affected instrumentation channels. The change provides branching notes to the Loss of Coolant Accident (LOCA) Time Delay Relay (TDR) Functions of LCO [limiting condition of operation] 3.3.5.1 from instrument channels of the primary and secondary containment isolation channels of LCO 3.3.6.1 and LCO 3.3.6.2 and the associated support features for the LOCA TDR function. Since these instruments affect multiple LCOs, this change will assure that operators implement the most restrictive Action and Completion Time when a channel becomes inoperable or is placed in the tripped condition. Providing this branching to the more restrictive Actions makes explicit what is currently required for Operability and has no impact on any previously evaluated accident.

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 does not impact any operational or physical aspect of WNP–2. The change only makes explicit the LCOs affected by the primary and secondary containment isolation instruments and the associated supported features for the LOCA TDR function.

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 provides branching notes to the LOCA TDR channels of LCO 3.3.5.1 from instrument channels of the primary and secondary containment isolation channels of LCO 3.3.6.1 and LCO 3.3.6.2 and provides notes for identifying associated support features for the LOCA TDR function. This change only makes explicit what is currently required for LCO 3.3.5.1 Functions 1c, 1d, 2c and 2d instrument channel Operability. This change will make explicit the most restrictive Action when an instrument sensor or channel becomes inoperable or is placed in the tripped condition, thereby, maintaining the margin of safety in accordance with the Technical Specifications.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 3, 2000.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) 3.7.12, "Control Room Emergency Ventilation System (CREVS)," ITS 5.6.2.12, "Ventilation Filter Testing Program (VFTP)," ITS 3.3.16, "Control Room Isolation—High Radiation," and ITS 3.7.18, "Control Complex Cooling System." The proposed ITS changes are based on the results of revised public and control room dose calculations for CR–3 design basis radiological accidents using an alternative source term (AST).

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously analyzed. The CR–3 Control Room Emergency Ventilation System (CREVS) and the Control Complex Habitability Envelope (CCHE) only function following the initiation of a design basis radiological accident. Therefore, the changes to the CREVS specification, the CREVS filter testing criteria, and the deletion of the requirement for control room isolation on high radiation proposed by this amendment will not increase the probability of any previously analyzed accident. The Control Complex Cooling System and Auxiliary Building Ventilation System are not initiators of any design basis accident. Therefore, the changes to the Control Complex Cooling System specification and the changes to the testing guidelines for the Auxiliary Building Ventilation System exhaust filters proposed by this amendment will not increase the probability of occurrence of any previously analyzed accident.

Revised dose calculations, which take into account the changes proposed by this amendment and the use of an AST, have been performed for the CR–3 design basis radiological accidents. The results of these revised calculations indicate that public and control room doses will not exceed the limits specified by 10 CFR 50.67 and Regulatory Guide 1.183. In addition, a comparison between results of the current public dose calculations and the revised public dose calculations indicate that the proposed changes will not result in a significant increase in predicted dose consequences for any of the analyzed accidents. Therefore, the proposed changes do not involve a significant increase in the consequences of any previously analyzed accident.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed.

Limiting the requirements for the Control Complex Cooling System and CREVS to be operable to Modes 1, 2, 3, and 4, and changing the Auxiliary Building Ventilation System exhaust filter testing guidelines do not result in changes to the design or operation of these systems. Although the other changes proposed by this amendment could affect the operation of the CREVS and CCHE following a design basis radiological accident, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. Therefore the proposed changes do not create the possibility of a new or different kind of accident from those previously analyzed.

3. Does not involve a significant reduction in the margin of safety.

The proposed changes to the control complex cooling specification do not affect the ability of the system to maintain control complex temperatures within safety-related equipment operability limits when the equipment is required. The results of revised control room dose calculations indicate that the proposed changes to the CREVS specification, the CREVS filter testing criteria, and removal of the CREVS actuation signal on high radiation will not affect the ability of the CREVS and CCHE to maintain control room doses less than required limits during design basis radiological accidents. The revised dose calculations also indicate that the Auxiliary Building Ventilation System exhaust filters are not required in order to maintain public or control room doses less than required limits; therefore the proposed changes to the testing requirements for these filters cannot adversely affect public or control room doses.

Based on the above, the revised technical specifications meet the same intent as the currently approved specifications. 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: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: June 7, 1999, as supplemented February 4, 2000.

Description of amendment request: The proposed amendment requests the staff to evaluate the integrity of the Kewaunee Reactor Pressure Vessel (RPV) circumferential beltline weld using a Master Curve-based methodology.

The licensee submitted a request for exemptions to 10 CFR 50.61, 10 CFR 50 Appendix G, and 10 CFR 50, Appendix H, to allow the use of the Master Curve-based methodology for calculating the RPV Reference Temperature for Pressurized Thermal Shock (RT_{PTS}) based on the fracture toughness data from irradiated pre-cracked Charpy V-notch specimen testing of Kewaunee and Maine Yankee surveillance welds. The Master Curve methodology is based on American Society for Mechanical Engineers (ASME) Code Case N–629 and American Society for Testing and Materials Standard (ASTM) E–1921. In its submittals, the licensee also requested a revision of the facility's Pressure-Temperature (P/T) limit curves.

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.

Failure of a reactor vessel is not an accident that has been previously evaluated. Design provisions ensure that this is not a credible event. Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Compliance with 10 CFR 50.61, 10 CFR 50 Appendix G and H, and application of ASME Code Case N–514, ASME Code Case N–588, and the exemption requested in Attachment 1 ensures that failure of a reactor vessel will not occur. The proposed changes do not impact the capability of the reactor coolant pressure boundary piping (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a LOCA.

The LTOP setpoint, LTOP system enabling temperature, and revised P/T limits reflected in proposed Figures TS 3.1–1 and TS 3.1–2 ensure that the Appendix G pressure/temperature limits are not exceeded, and therefore, ensure that RCS integrity is maintained. The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, construction standards, or setpoints. The reactor coolant system full power operating pressure (2235 psig) is not being changed by this proposed amendment. The LTOP valve setpoint remains at ≤500 psig. The LTOP enabling temperature based on Figure TS 3.1–2 is 200°F and is consistent with ASME Code Case N–514 guidance of RT_{NDT} + 50°F. The LTOP enabling temperature is not changed by this amendment. The allowable combination of Appendix G pressure and temperature for the cooldown limits is marginally greater than the current limits. The combination of slightly greater allowable Appendix G pressure and temperature limits and low enabling temperature produces an adequate operating window. An adequate operating window reduces the likelihood of inadvertently lifting the LTOP relief valve while maneuvering the plant through the knee of the P–T curve during startup and shutdown. The probability of an LTOP event occurring is independent of the pressure-temperature limits for the RCS pressure boundary and enabling temperature. Therefore, the probability of a LTOP event is not increased.

The revised heatup and cooldown limit curves and corresponding LTOP enabling temperature were developed using test results from unirradiated and/or irradiated specimens that represent the KNPP reactor vessel beltline circumferential weld, closure head flange, and intermediate forging. The circumferential beltline weld and intermediate forging are the most limiting materials in the reactor coolant pressure boundary. These materials are limiting due to the effects of neutron irradiation which cause the flow properties to increase and the toughness to decrease. The circumferential beltline weld is the controlling material for evaluation of pressurized thermal shock. With NRC approval to use Code Case N–588 and the exemption requested in Attachment 1, the reactor vessel intermediate forging and head flange become the limiting and controlling materials for development of the Appendix G limit curves and corresponding LTOP system enabling temperature. 10 CFR 50, Appendix G states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. Fracture toughness, drop weight, and Charpy V-notch testing of the 1P3571 weld metal and drop weight, and Charpy V-notch testing of the intermediate forging material has been performed. The results of those tests have been used for derivation of the revised PTS assessment, the proposed Appendix G heatup and cooldown limit curves, and the corresponding LTOP system enabling temperature. The revised limit curves and corresponding LTOP enabling temperature have been developed using accepted engineering practices. The evaluations were performed in accordance with methods derived from the ASME Boiler and Pressure Vessel Code, criteria set forth in NRC Regulatory Standard Review Plan 5.3.2, and 10 CFR 50.61. The revised heatup and cooldown limit curves and corresponding LTOP enabling temperature ensures adequate fracture toughness for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary. These limit curves provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection [corresponding to isothermal events during low temperature operations (i.e., ≤200°F)], thus ensuring the integrity of the reactor coolant pressure boundary.

The changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the changes do not affect any fission product barrier. The changes do not degrade or prevent the response of the LTOP relief valve or other safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the consequences of an accident previously evaluated will not be increased.

Thus, operation of KNPP in accordance with the PA [proposed amendment] does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Application of ASME Code Case N–514, ASME Code Case N–588, and the exemption requested in Attachment 1 ensures that failure of a reactor vessel will not occur. Therefore, a failure of the reactor vessel can still be considered incredible.

The proposed heatup and cooldown limit curves have been constructed by combining the most conservative pressure-temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. Use of the proposed curves, does not modify the reactor coolant system pressure boundary, nor make any physical changes to the LTOP setpoint or design. Proposed Figures TS 3.1–1 and TS 3.1–2 were prepared in accordance with regulatory and code requirements and were derived using conservative material property basis and neutron exposure projections thru 33 EFPY. Therefore, the proposed heatup and cooldown curves and LTOP limits will continue to protect the reactor vessel from failure.

The LTOP system enabling temperature and the proposed Appendix G pressure temperature limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. The changes do not cause the initiation of any accident nor create any new credible limiting failure for safety-related systems and components. The changes do not result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than previously evaluated. The changes do not have any adverse effect on the ability of the safety-related systems to perform their intended safety functions.

The proposed changes do not make physical changes to the plant or create new failure modes. Thus, the PA [proposed amendment] does 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 Appendix G pressure temperature limitations and corresponding LTOP enabling temperature were prepared using methods derived from the ASME Boiler and Pressure Vessel Code, including ASME Code Cases N–514 , N–588, and N–629.

Inherent conservatism in the P/T limits resulting from these documents is described in the Safety Evaluation.

Alternative methodologies to the safety margins required by Appendix G to 10 CFR Part 50 have been developed by the ASME Working Group on Operating Plant Criteria. Three of these methodologies are contained in ASME Code Cases N–514, N–588, and N–629.

Code Case N–514 provides criteria to determine pressure limits during LTOP events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of the relief valve used for LTOP. Specifically, the ASME Code Case N–514 allows determination of the setpoint for LTOP events such that the maximum pressure in the vessel would not exceed 110% of the P/T limits of the existing ASME Appendix G; and redefines the enabling temperature at a coolant temperature less than 200 °F or a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater. Code Case N–514, ''Low Temperature Overpressure Protection,'' has been approved by the ASME Code Committee but not yet approved for use in Regulatory Guides 1.147, 1.85, or 1.84. The content of this Code Case has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. It is expected that the next revision of 10 CFR 50.55a will endorse the 1993 Addenda and Appendix G of Section XI. Code Case N–514 is not in conflict with 10 CFR 50.61 and therefore has been used to establish the LTOP system enabling temperature; the provision for exceeding 110% of the Appendix G limits has not been incorporated in PA [proposed amendment] 160. The NRC previously approved use of Code Case N–514 for determination of the LTOP enabling temperature in Reference 6.

Code Case N–588 provides benefits in terms of calculating pressure-temperature limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The NRC previously approved use of Code Case N–588 for use at KNPP in references 4 and 5.

In support of this PA [proposed amendment], WPSC used fracture toughness results representing the beltline weld metal that were irradiated to EOL and in excess of EOLE fluence. The fracture toughness results were analyzed as described under Case #6 in WCAP–15075 and ASME Code Case N–629 for determining the EOL and EOLE indexing reference temperature values. Attachment 1 to this letter provides information to support NRC approval to use the weld metal fracture toughness results along with the methodology presented in WCAP–15075 for the KNPP PTS evaluation. The KNPP application of the methodology presented in WCAP–15075, identified as Case #6, incorporates the following additional margins beyond that recommended in ASTM E1921–97:

(a) A delta value of 17 °F is added to T_0 to ensure that the margin in the KNPP application is at least as conservative as the margin associated with the most limiting HSST–02 plate material.

(b) An additional margin of 18 °F has been added to the above 17 °F to be consistent with the ASME Code Case N–629, and align the KNPP lead plant application with current consensus of the technical community regarding the best use of fracture toughness based indexing reference temperature data.

(c) A 2 σ value of 16 °F and 24 °F is added to account for RT_{T_0} measurement uncertainty for EOL and EOLE, respectively.

(d) A value of (+)35 °F and (−)32 °F accounts for heat uncertainty between the KNPP and Maine Yankee surveillance capsule specimens for EOL and EOLE, respectively.

Fracture toughness testing of irradiated 1P3571 weld metal, performed in accordance with ASTM E1921–97 and application of ASME Code Case N–629 along with the methods in WCAP–15075, indicate that the end of life indexing reference temperature is 234 °F. This fracture toughness generated EOL indexing reference temperature value includes a margin of 34 °F (18 °F + 16 °F). The fracture toughness generated indexing reference temperature value (234 °F) is lower than the ART value (277 °F) predicted by the Charpy V-notch and Drop Weight methodology. Both methodologies predict end of life indexing reference temperature values that are below the pressurized thermal shock screening criteria (300 °F).

Use of the methodology set forth in the ASME Boiler and Pressure Vessel Code, NRC Regulatory Standard Review Plan 5.3.2., WCAP–15075, 10 CFR 50.61, and 10 CFR 50 Appendices G and H ensures that proper limits and safety factors are maintained. Thus, the PA [proposed amendment] does not involve a significant reduction in the margin of safety.

The revised heatup and cooldown limit curves and corresponding LTOP system enabling temperature were prepared using fracture toughness, drop weight and Charpy V-notch data for the beltline weld material; drop weight and Charpy V-notch data for the closure head flange and intermediated forging material; along with practices described herein and methods derived from the ASME Boiler and Pressure Vessel Code and 10 CFR 50.61. The safety factors and margins used in the development of the limit curves and LTOP system enabling temperature meet the criteria set forth by these documents. Application of low leakage core designs decreases the rate of shift in transition temperature from ductile to nonductile behavior. The revised limit curves and corresponding LTOP enabling temperature provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection [corresponding to isothermal events during low temperature operations (i.e., ≤200 °F)]. With the preparation of the revised limit curves in accordance with the latest criteria and guidance, this proposed amendment ensures that proper limits and safety factors are maintained.

Thus, the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, the proposed amendment does not represent a significant decrease in the margin of safety. As shown in Attachment 1 [in the proposed amendment], a loss of reactor vessel integrity is still incredible. Furthermore, the LTOP setpoint and enabling temperature will continue to protect the reactor coolant system during low temperature operation.

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.

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

Date of amendment request: September 7, 2000.

Description of amendment request: The proposed amendment to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs) would reflect a modification planned for refueling outage (RO) 11, scheduled to begin in May of 2001. The modification will automatically close, on a safety injection signal, the existing main feedwater inlet isolation valves (MFIIVs) and the main feedwater low flow bypass inlet isolation valves.

Basis for proposed no significant hazards consideration determination:

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

Operation of the Indian Point 3 plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

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

The proposed TS change reflects a planned modification to automatically isolate main feedwater on a safety injection signal using the motor operated Main Feedwater Inlet Isolation Valves (MFIIVs) and MF [main feedwater] low flow bypass inlet isolation valves. These non-safety valves will be incorporated into the IST [inservice testing] program as augmented components and included in the Generic Letter 89–10 program for motor operated valves. The modification will not relocate the safety injection signal from the Main Boiler Feedpump Discharge Valves (MBFPDVs) but closure will no longer be assumed in analyses. The modification is based on current design function for the feedwater isolation following a main steam line break inside containment accomplished by MBFPDVs. The TS changes add a limiting condition for operation, required action statements with completion times and surveillance requirements that are the same as those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG–1432. The plant core reload analysis will assume that the modification is complete (this eliminates the continued addition of the feedwater between the MFIIVS and associated bypass valves and the MBFPDVs) and demonstrate that a shutdown margin of 1.3% is acceptable and that no boron concentration needs to be assumed in the safety injection lines. The proposed changes cannot affect the probability of an accident occurring since they reflect a change in plant design consistent with current design which is not an accident initiator. The proposed changes cannot increase the consequences of postulated accidents since they reflect a change in plant design that will mitigate the effects of feedwater to a faulted steam generator for a main steam line break inside containment and restore past analytical assumptions regarding a 1.3% shutdown margin and no boron in the safety injection lines.

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

The proposed TS change reflects a planned modification to automatically isolate main feedwater on a safety injection signal using the motor operated Main Feedwater Inlet Isolation Valves (MFIIVs) and MF low flow bypass inlet isolation valves. These non-safety valves will be incorporated into the IST program as augmented components and included in the Generic Letter 89–10 program for motor operated valves. The modification will not relocate the safety injection signal from the Main Boiler Feedpump Discharge Valves (MBFPDVs) but closure will no longer be assumed in analyses. The modification is based on current design function for the feedwater isolation following a main steam line break inside containment accomplished by MBFPDVs. The TS changes add a limiting condition for operation, required action statements with completion times and surveillance requirements that are the same as those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG–1432. The proposed TS changes do not create the possibility of a new or different type of accident from those previously evaluated since they reflect a design change that will accomplish the same feedwater isolation function as previously done by the MBFPDVs with no change to the manner in which the feedwater system operates.

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

The proposed TS change reflects a planned modification to automatically isolate main feedwater on a safety injection signal using the motor operated Main Feedwater Inlet Isolation Valves (MFIIVs) and MF low flow bypass inlet isolation valves. These non-safety valves will be incorporated into the IST program as augmented components and included in the Generic Letter 89–10 program for motor operated valves. The modification will not relocate the safety injection signal from the Main Boiler Feedpump Discharge Valves (MBFPDVs) but closure will no longer be assumed in analyses. The modification is based on current design function for the feedwater isolation following a main steam line break inside containment accomplished by MBFPDVs. The TS changes add a limiting condition for operation, required action statements with completion times and surveillance requirements that are the same as those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG–1432. The plant core reload analysis will assume that the modification is complete (this eliminates the continued addition of the feedwater between the MFIIVS and associated bypass valves and the MBFPDVs) and demonstrate that a shutdown margin of 1.3% is acceptable and that no boron concentration needs to be assumed in the safety injection lines. The proposed TS change cannot involve a significant reduction in the margin of safety since it is based upon a modification that will restore the margin of safety with respect to feedwater addition, shutdown margin and core boration for a main steam line break inside containment to the previously analyzed condition. This assumes that loading of the valves on the emergency diesel generators will not affect the emergency diesel generators margin.

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. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: September 7, 2000.

Description of amendment request: The proposed amendment to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs) would extend allowed outage times (AOTs) on a one-time basis, before May 31, 2002, to allow for replacement of the 31 and 32 station batteries while the plant is on line. The proposed amendment also removes an expired footnote regarding repairs to the 32 diesel fuel oil tank.

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

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

No. The proposed AOT extension does not involve a significant increase in the probability or consequences of an accident previously evaluated. During the replacement of the existing station batteries, a temporary battery will provide the same function as the Exide batteries being removed. Even though this temporary battery will not meet seismic, seismic interaction or security requirements, due to its location on the 53-ft elevation of the Turbine Building, it is qualified as safety related in all other respects. The 125 VDC EDS [electrical distribution system] is normally supplied by the associated 480 VAC bus through a Battery Charger. The essential function of 31, 32 and 33 station battery is to supply DC control power necessary to start and load the associated EDG [emergency diesel generator]. Once the EDGs are on line, the 125 VDC EDS will be supplied via the battery charger. However, the station batteries have been sized to carry shutdown loads for a period of two hours without battery terminal voltage falling below its minimum required voltage following a plant trip that includes a loss of all AC power. This provides additional assurance that the critical DC loads are available in the event of a loss of the battery charger. During the 10-day AOT, when the temporary battery and the associated battery charger are supporting the 125 VDC bus, the ability of that ESF [engineered safety feature] DC power panel to mitigate an event/accident remains unchanged except for its ability to cope with a seismic, seismic interaction or security event. However, the probability of these types of events concurrent with the 10-day AOT is very small. During these types of events, one ESF DC power panel may be compromised, however IP3 has adequate 125 VDC power available in the form of two other ESF train DC power panels to mitigate all DBAs. The postulated loss of one ESF DC power panel is bounded by the loss of an entire ESF electrical train, a condition which the plant is currently evaluated to withstand. Based upon the above, the overall design, function and operation of the 125 VDC EDS and equipment has not been significantly modified by the proposed changes. The proposed changes do not affect accident initiators or precursors, nor do they alter the design assumptions for the systems or components used to mitigate the consequences of an accident as analyzed in Chapter 14 of the IP3 USFAR [UFSAR] [updated final safety analysis report], except for one of the three trains of DC power. The remaining DC power trains can mitigate a DBA [design-basis accident]. Therefore, the proposed one-time AOT extension TS amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. During the replacement of the existing station batteries, a temporary battery will provide the same function as the batteries being removed. Even though this temporary battery does not meet all design requirements of a seismic, seismic interaction or security event it possesses adequate capacity to fulfill the safety related requirements of supplying necessary power to the associated 125 VDC bus under most conditions. Because the temporary battery will perform like the station battery that is currently installed, and will be connected and used in the same way as a backup power supply to the DC bus, no new electrical or functional failure modes are created. The temporary battery will be located in the turbine building, which is non-seismic and a non-vital area. The temporary battery will not be placed into seismically mounted racks. Thus, a seismic failure of this temporary battery is possible. Since the temporary battery is located in the turbine building the potential for battery failure to initiate an accident is not present. The failure of the temporary battery cannot create a different response from any previously postulated accident. Due to the location of the main turbine-generator in relationship to the temporary battery, it is not likely that a turbine missile would strike the battery. Likewise, an unmitigated Steam Line Break accident outside the VC would be interrupted by successful closure of all MSIVs [main steam isolation valves] thereby leaving the battery and the associated DC bus intact and available. This MSIV closure would occur before any potential steam line break impacting the battery on the Turbine deck ensuring necessary DC power to the MSIVs when needed. Also, any affects of postulated severe weather on the turbine building have been evaluated and do not impede the ability of the remaining DC subsystems to perform their intended safety function. The remaining 125 VDC EDS and its equipment will continue to perform the same function and be operated in the same fashion. The proposed changes do not introduce any new accident initiators or precursors, or any new design assumptions for those systems or components used to mitigate the consequences of an accident. Therefore, the possibility of a new or different kind of accident from any previously evaluated has not been created. Thus, the proposed one-time AOT extension TS amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. During the replacement of the existing station batteries, a temporary safety related battery will perform the same function as the battery being removed. Even though this battery is not seismically mounted, not in a seismically qualified building, nor in a vital area of the plant it is qualified as a safety related battery in all other respects.

This battery is virtually identical to the safety related station battery that is already installed. It possesses adequate capacity to fulfill the requirements of the associated 125 VDC bus. The proposed replacement activity will not prevent the plant from mitigating a DBA during events that result in the loss of the temporary battery. In these cases, the remaining DC power supporting the design mitigation capability will be maintained. Due to the limited duration of the activity, the very low probability of a seismic or other seismic interaction event over this limited AOT period and the planned implementing contingency actions, a significant reduction in the margin of safety will not result. The associated DC bus will always be supplied with both a temporary battery and a battery charger at all times. The inherent design conservatism of the 125 VDC system and its equipment has not been significantly altered; only the degree of redundancy is not fully qualified. The 125 VDC EDS and its equipment will continue to be operated with the same degree of conservatism. Accordingly, there is no significant reduction in the margin of safety.

Therefore, based upon the above evaluation, the Authority has concluded that these changes involve no 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. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: September 7, 2000.

Description of amendment request: The proposed amendment to the Indian Point Nuclear Generating Unit No. 3 Technical Specifications would extend the surveillance frequency from 720 hours to 1440 hours for the Fuel Storage Building Emergency Ventilation system.

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

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

Response: The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Extending the surveillance frequency from 720 hours to 1440 hours for the Fuel Storage Building Emergency Ventilation (FSBEV) System charcoal and HEPA [High Efficiency Particulate Adsorbers] adsorbers does not involve any modifications to the plant, will not require changes to how the plant is operated nor will it affect the operation of the plant. Filter systems are not initiators of accidents, and therefore extending the filter surveillance frequency will not increase the probability of an accident. The way the filters perform will not be changed by extending the surveillance frequency. In addition, it is reasonable to expect satisfactory filter performance at this extended frequency based on past surveillance results. Hence, there is no change in the assumptions of an accident. Therefore, this change will not increase the consequences of an accident previously evaluated.

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

Response: The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Extending the surveillance frequency from 720 hours to 1440 hours for the FSBEV charcoal and HEPA adsorbers does not involve any modifications to the plant, will not require changes to how the plant is operated nor will it affect the operation of the plant. Therefore, extending the surveillance frequency will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The proposed license amendment does not involve a significant reduction in a margin of safety. Extending the surveillance frequency from 720 hours to 1440 hours for the FSBEV charcoal and HEPA adsorbers does not change the TS required methyl iodine efficiency removal requirement of >90% that ensures a safety factor of at least 2. This change is acceptable because it is reasonable to expect satisfactory filter performance at this extended frequency based on past surveillance results, hence it is reasonable to expect that the additional 720 hours before testing will not result in the safety factor being diminished. Thus, the proposed change would 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. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni.

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 amendment request: September 26, 2000, as supplemented on October 6, 2000.

Description of amendment request: The proposed change would amend the Salem Nuclear Generating Station (Salem) Unit Nos. 1 and 2 Technical Specifications (TSs) to increase the as-found set point tolerance for the Pressurizer Safety Valves (PSV) from ±1% to ±3%; increase the as-found set point tolerance for the Main Steam Safety Valves (MSSV) from ±1% to ±3%; change the required action for inoperable MSSVs to require a reduction in power based upon the number of inoperable MSSVs, as opposed to the current requirement to reduce the Power Range Neutron Flux High trip setpoint; and remove specifications and references related to plant operation with three Reactor Coolant System loops. The associated TS Bases sections will also be amended to reflect the TS changes.

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

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

Changing the pressurizer and main steam safety relief valve lift setpoint tolerance from ±1% to ±3% does not significantly increase the probability of any accident previously evaluated. The only events initiated by the opening of these safety valves are the accidental depressurization of the Reactor Coolant System and accidental depressurization of the Main Steam System. These events are a result of an inadvertent lifting of these valves and do not depend on the safety valve lift setpoint or tolerance. Therefore, the likelihood that either of these events will occur has not been increased.

[Analyses associated with the limiting overpressurization transients (Loss of External Electrical Load and/or Turbine Trip, and Single Reactor Coolant Pump Locked Rotor) have been performed that demonstrate that increasing the Pressurizer Safety Valve and Main Steam Safety valve lift setpoint tolerance to ±3% would result in primary and secondary side pressure responses less than the acceptance criteria of 110% of the design pressure. Therefore, since the proposed setpoint tolerance increase would not adversely impact current accident analysis assumptions, the proposed change would not result in an increase in consequences of an accident previously evaluated.]

For operation with inoperable main steam safety valves, changing the required action from a reduction of the power range high neutron flux trip setpoint to a reduction of the allowable reactor power level will not increase the consequences of any accident. With inoperable Main Steam Safety Valves, the Loss of External Electrical Load and/or Turbine Trip event becomes limiting in terms of secondary side pressurization. The high flux trip does not provide any mitigation for this event. Other events limiting at power, that require the power range trip for mitigation, assume a safety analysis trip setpoint of 118% (based on a nominal trip setpoint of 109%) regardless of the initial power level. Therefore, the proposed change does not impact any of the accident analysis assumptions.

The current Salem licensing basis for the Spurious Activation of the Safety Injection System credits operator action to unblock a pressurizer Power Operated Relief Valve prior to the water solid pressurizer reaching the safety valve lift setpoint. The analyses that determined the time at which the safety valve would reach its pressure setpoint covered the −3% tolerance. Since this would conservatively result in the earliest opening time, there was no need to consider the positive side of the tolerance. The results of the analyses indicate that the allowable operator action time is sufficient, such that water relief occurs through the Power Operated Relief Valves and not through the Pressurizer Safety Valves. As such the consequences of this event have not changed as a result of the proposed change.

Increasing the Main Steam Safety Valve lift setting tolerance may result in increased secondary side backpressure for the Auxiliary Feedwater Pumps. However, analyses have demonstrated that with the elevated backpressures that could result from increasing the Main Steam Safety Valve setpoint upper tolerance to +3%, the Auxiliary Feedwater Pumps would still provide [greater than the minimum] flow required to mitigate events in which normal feedwater is not available, a Loss of Normal Feedwater and a Loss of Offsite Power to Station Auxiliaries.

In terms of radiological consequences, the current design and licensing basis analyses that include steaming through the Main Steam Safety Valves bound the proposed lift setpoint tolerance change.

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

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

The proposal will result in a change in the allowed Pressurizer Safety Valve and Main Steam Safety Valve lift setpoint tolerance range. No physical changes to these valves or to their nominal lift setpoint is required. These valves are assumed to malfunction only as the initiator for the accidental depressurization of the Reactor Coolant System or Main Steam System. An increased lift setpoint tolerance range does not change the assumption of these depressurization events nor create a new type of event.

Requiring a reduction in reactor thermal power in the event of inoperable Main Steam Safety Valves is consistent with the analysis methodology. Initiation of any Salem UFSAR [Updated Final Safety Analysis Report] analyzed event at a power level less than full power is bounded by those events analyzed at full power, or specifically analyzed at the limiting power level, and does not constitute a new or different kind of accident. Also, no changes are being made to the power range high flux trip setpoint that will make it inconsistent with any analytical assumption.

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

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

Analyses performed demonstrate that the proposed increase in the Pressurizer Safety Valve and Main Steam Safety Valve lift pressure setpoint tolerance from ±1% to ±3% will provide acceptable primary and secondary side pressure responses to the anticipated operational occurrences and design basis accidents. The limiting overpressurization transients, Loss of External Electrical Load and/or Turbine Trip, and Single Reactor Coolant Pump Locked Rotor, stay well within the acceptance criteria of 110% of the design pressure.

For operation with inoperable Main Steam Safety Valves, requiring a reduction in reactor thermal power is consistent with the accident analysis. The current requirement to reduce the power range high neutron flux trip setpoint [does not reduce the] margin of safety since this trip does not provide any mitigation for the limiting secondary system pressurization event, Loss of External Electrical Load and/or Turbine Trip with inoperable Main Steam Safety Valves.

The current licensing basis for the Spurious Activation of the Safety Injection System credits operator action to unblock a pressurizer Power Operated Relief Valve prior to the water solid pressurizer reaching the Pressurizer Safety Valve lift setpoint. As the Pressurizer Safety Valves are not designed for water relief, failure to unblock a Power Operated Relief Valve before reaching the Pressurizer Safety Valve lift setpoint would result in water relief and likely failure of the Pressurizer Safety Valve to reseat. This condition would escalate the Spurious Activation of the Safety Injection System (Condition II event) into a small break Loss Of Coolant Accident (Condition III event). The analyses that determined the time at which primary system pressure would reach the Pressurizer Safety Valve setpoint bound the −3% tolerance. The results of the analyses indicate that the allowable operator action time is sufficient, such that water relief occurs through the Power Operated Relief Valves and not through the Pressurizer Safety Valves. Since the Pressurizer Safety Valve would not fail due to water relief, there is no reduction in the margin of safety for this event.

Increasing the Main Steam Safety Valve lift setting tolerance may result in increased secondary side backpressure for the Auxiliary Feedwater System. However, analyses have demonstrated that under degraded Auxiliary Feedwater Pump performance, and with secondary side backpressure corresponding to 103% of the lowest Main Steam Safety Valve setpoint, the Auxiliary Feedwater System can provide [greater than the minimum] flow required to mitigate those events where normal feedwater is not available, a Loss of Normal Feedwater and a Loss of Offsite Power to Station Auxiliaries.

Therefore the proposed changes to the Technical Specifications 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment requests: October 6, 2000 (PCN–518).

Description of amendment requests: The amendment application proposes to revise the San Onofre Nuclear Generating Station, Units 2 and 3, Technical Specification (TS) 3.7.11, ''Control Room Emergency Air Cleanup System (CREACUS)'' consistent with generic industry changes recently approved by the U.S. Nuclear Regulatory Commission (NRC) document Technical Specification Task Force (TSTF)–287. The proposed amendments would allow up to 24 hours to restore the Control Room Pressure Boundary (CRPB) to operable status when two CREACUS trains are inoperable due to an inoperable CRPB in MODE 1, 2, 3, or 4. In addition, a Limiting Condition for Operation note would be added to allow intermittent opening of the CRPB under administrative controls without entering the Actions.

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

Operation of the facility in accordance with the proposed amendments does not:

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

Response: No.

The Control Room Area Ventilation System and Control Room boundary are not assumed to be an initiator of any analyzed accident; they are provided to minimize doses to the control room operators during an accident. Therefore, these proposed changes have no impact on the probability of occurrence of any previously analyzed accident.

The proposed changes also have no impact on offsite dose consequences. The control room ventilation system and control room boundary provide protection for control room personnel and do not mitigate radiological effluents released offsite. With the control room boundary inoperable and not pressurized, the accident analyses assume unfiltered air would enter the control room and operator doses would be significantly increased. Conservative accident analysis assumptions do not take credit for available compensatory measures to mitigate operator dose. Compensatory measures include the supply of protective clothing, and self contained breathing apparatus adequate for at least nine persons within the control room envelope.

Additionally, for cases where the control room boundary is opened under administrative control, appropriate administrative measures ensure the boundary can be rapidly restored. Based on the compensatory measures available to the control room operator to minimize dose (to be consistent with the intent of General Design Criterion 19), the administrative controls required to rapidly restore an opened boundary, and considering the low probability of an event occurring in this short time period, the consequences are not considered to be significantly increased. Operators maintain the ability to mitigate a design basis event.

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

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

Response: No.

No changes are being made to actual plant hardware which will result in any new accident causal mechanisms. Therefore, no new accident causal mechanisms will be generated.

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

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

Response: No.

Margin of safety is related to the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be degraded by the proposed changes. The Control Room Ventilation System and control room boundary provide a protected environment for the control room operators during analyzed events. The proposed change would allow the boundary to be degraded for a limited period of time. However, administrative controls would be in place to rapidly restore an opened boundary and existing compensatory measures (e.g., protective clothing and self contained breathing apparatus) would be implemented to minimize operator dose. Therefore, it is expected that operators would maintain the ability to mitigate design basis events and none of the fission product barriers would be affected by this change.

Therefore, this 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 requests involve 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: Stephen Dembek.

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 amendment request: October 6, 2000.

Description of amendment request: The proposed amendment would amend each of the three units' Technical Specifications (TS) to adopt Technical Specifications Task Force (TSTF) change No. 318, Revision 0 (TSTF–318). TSTF–318 provides a 7-day action period and completion time in the event of inoperability of one of the two low pressure coolant injection (LPCI) pumps in each of the two emergency core cooling system (ECCS) divisions.

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 new Condition of one LPCI pump in each LPCI injection subsystem being inoperable is more reliable than the current Limiting Condition for Operation which allows 2 LPCI pumps in one ECCS subsystem to be inoperable for 7 days. Also, the LPCI mode of the Residual Heat Removal system is not assumed to be initiator of any analyzed event. Therefore, the proposed amendment does not involve a significant 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 change does not involve a physical alteration of the plant, add any new equipment or require any existing equipment to be operated in a manner different from the present design. The proposed change will not impose any new or eliminate any existing requirements. Therefore, the proposed amendment 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 proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. The proposed new Condition for one LPCI pump in each LPCI injection subsystem represents a more reliable configuration than the existing LCO which allows two LPCI pumps in one ECCS subsystem to be inoperable for 7 days. For these reasons, the proposed amendment 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET I0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: June 16, 2000, as supplemented by letter dated September 27, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7 and TS Tables 3.7–1, 3.7–2, 3.7–3, and 4.1–1. The proposed changes would: (a) revise the surveillance frequency for Reactor Protection System and Engineered Safety Features Actuation System analog channels from monthly to quarterly; (b) decrease the frequency for most permissives to a refueling interval; (c) increase the time allowed to perform maintenance on an inoperable instrument channel; and (d) revise associated action statements consistent with NUREG–1431.

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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) Technical Specification changes for the Surry Units 1 and 2 and determined that a significant hazards consideration is not involved. In support of this conclusion, the following evaluation is provided.

Criterion 1—Operation of Surry Units 1 and 2 in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The determination that the results of the proposed changes remain within acceptable criteria was established in the SER(s) [Safety Evaluation Report(s)] prepared for WCAP–10271, WCAP–10271 Supplement 1, WCAP–10271 Supplement 2, WCAP–10271 Supplement 2, Revision 1 and WCAP–14333 issued by letters dated February 21, 1985, February 22, 1989, April 30, 1998, and July 15, 1998.

Implementation of the proposed changes is expected to result in an increase in total RPS and ESFAS yearly unavailability. The proposed changes have been shown to result in a small increase in the core damage frequency (CDF) due to the combined effects of increased RPS and ESFAS unavailability and reduced inadvertent reactor trips.

The values determined by the WOG [Westinghouse Owners Group] and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analyses for the NRC Staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered acceptable. The analysis performed by the WOG and presented in the WCAP included changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers. The overall increase in the CDF, including the changes to the surveillance frequencies for the automatic actuation logic and actuation relays and the reactor trip and bypass breakers, was approximately 6 percent. However, even with this increase, the overall CDF remains lower than the NRC safety goal of 10 E–4/reactor year.

Changes to surveillance test frequencies for the RPS and ESFAS interlocks do not represent a significant reduction in testing. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the annunciator status light. The surveillance as currently required only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability is verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. Therefore, the change in the interlock surveillance requirement to at least once every 18 months does not represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the RPS and ESFAS.

For the additional relaxations in WCAP–14333, the WOG evaluated the impact of the additional relaxation of allowed outage times and completion times, and action statements on core damage frequency. The change in core damage frequency is 3.1 percent for those plants with two out of three logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP–10271 and its supplements. This analysis calculates a significantly lower increase in core damage frequency than the WCAP–10271 analysis calculated. This can be attributed to more realistic maintenance intervals used in the current analysis and crediting the AMSAC [ATWS (anticipated transient without scram) mitigating system actuation circuitry] system as an alternative method of initiating the auxiliary feedwater pumps. Therefore, the overall increase in CDF is estimated to be 3.1% for the proposed changes per the generic Westinghouse analysis.

The NRC performed an independent evaluation of the impact on core damage frequency (CDF) and large early release fraction (LERF). The results of the staff's review indicate that the increase in core damage frequency is small (approximately 3.2%) and the large early release fraction would increase by only 4 percent for 2 out of 3 logic schemes that have not implemented the proposed surveillance test interval, allowed outage times, and completion times evaluated in WCAP–10271 and its supplements. Further, the absolute values for CDF still remain within NRC safety goals.

Therefore, the proposed changes do not result in a significant increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS and ESFAS but does not alter the manner in which protection is afforded or the manner in which limiting criteria are established.

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

The proposed changes do not result in a change in the manner in which the RPS or ESFAS provide plant protection. No change is being made which alters the functioning of the RPS or ESFAS (other than in a test mode). Rather the likelihood or probability of the RPS or ESFAS functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident as defined in the Safety Analysis Report.

The proposed changes do not involve hardware changes. Some existing instrumentation is designed to be tested in bypass and current Technical Specifications allow testing in bypass. Testing in bypass is also recognized by IEEE [Institute of Electrical and Electronics Engineers] Standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore, since the other proposed changes do not alter the physical operation or functioning of the RPS or ESFAS, the possibility of a new or different kind of accident from any previously evaluated has not been created.

Criterion 3—The proposed license amendment does not involve a significant reduction in a margin of safety.

The proposed changes do not alter the safety limits, limiting safety system setpoints or limiting conditions for operation. The RPS and ESFAS analog instrumentation remain operable to mitigate as assumed in the accident analysis. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act.

Implementation of the proposed changes is expected to result in an overall improvement in safety by less frequent testing of the RPS and ESFAS analog instruments and will result in less inadvertent reactor trips and actuation of Engineered Safety Features components.

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

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

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

NRC Section Chief: Richard L. Emch.

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).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: August 25, 2000, as supplemented September 21, October 14, and October 25, 2000.

Brief description of amendment: The amendment revises the reactor vessel pressure-temperature limits.

Date of issuance: October 31, 2000.

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

Amendment No.: 134.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 19, 2000 (65 FR 56598).

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

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

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: July 27, 2000, as supplemented October 5, 2000.

Brief description of amendment: The amendment revises the Safety Limit Minimum Critical Power Ratio.

Date of issuance: November 3, 2000.

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

Amendment No.: 135.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* August 23, 2000 (65 FR 51348).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 2000.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: March 7, as supplemented on April 21, June 14, and September 15, 2000.

Brief description of amendment: The proposed amendment revised the Technical Specifications to revise the surveillance requirements from once per refueling interval for each excess flow check valve (EFCV) to testing a representative sample of EFCVs once per 24 months.

Date of Issuance: October 25, 2000.

Effective date: October 25, 2000 and shall be implemented within 30 days of issuance.

Amendment No.: 216.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* August 23, 2000 (65 FR 51354).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 7, 2000, as supplemented June 14 and September 11, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.7.6, ''Control Room Emergency Filtration System,'' TS 3/4.7.7, ''Reactor Auxiliary Building Emergency Exhaust System,'' and TS 3/4.9.12, ''Fuel Handling Building Emergency Exhaust System.'' Specifically, these TS have been revised to provide an action when the Control Room Emergency Filtration System or Reactor Auxiliary Building Emergency Exhaust System ventilation boundary is inoperable, and a note that allows an applicable ventilation boundary to be open intermittently under administrative controls. The associated TS Bases are also being changed in accordance with the amendment. In addition, TS 3/4.3.3.1, ''Radiation Monitoring for Plant Operations,'' has been modified to provide consistency between the applicability of the Control Room Emergency Filtration System and the radiation monitors that initiate a control room isolation signal.

Date of issuance: October 30, 2000.

Effective date: October 30, 2000.

Amendment No.: 102.

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

Date of initial notice in ***Federal Register****:* May 3, 2000 (65 FR 25762). The supplemental letters dated June 14 and September 11, 2000, contained clarifying information only, did not expand the application beyond the scope of the initial notice, and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 21, 2000.

Brief description of amendments: The amendments revised the condensate storage tank (CST) low-level setpoint to prevent entrainment of air in the high pressure coolant injection (HPCI) pump suction line when taking suction from the CST. The amendments also revised the surveillance requirements for the CST level instruments.

Date of issuance: October 31, 2000.

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

Amendment Nos.: 182 and 177.

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

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15376).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 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: November 23, 1999, as supplemented by letter dated September 6, 2000.

Brief description of amendments: The amendments revised the Technical Specifications 5.5.11—Ventilation Filter Testing Program, which provides the test requirements for charcoal filters, to assure compliance with the requirements of American Society for Testing and Materials D3803–1989.

Date of issuance: November 2, 2000.

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

Amendment Nos.: 196/177.

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

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15377).

The supplement dated September 6, 2000, provided clarifying information that did not change the scope of the November 23, 1999, application and initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 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: August 10, 2000.

Brief description of amendment: The amendment revised the Technical Specifications to allow an alternate storage configuration of fuel assemblies adjacent to the walls within Region I of the spent fuel pool.

Date of issuance: October 24, 2000.

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

Amendment No.: 224.

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

Date of initial notice in ***Federal Register:*** September 6, 2000 (65 FR 54086).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 7, 2000, as supplemented October 2 and 4, 2000.

Brief description of amendments: The pressure-temperature limits specified in Technical Specification (TS) 3.4.9.1 and Figures 3.4–2, 3.4–3 have been modified, Figure 3.4–4 deleted, and the Cold Overpressure Mitigation System (COMS) requirements have been changed. The COMS is the Westinghouse version of the Low Temperature Overpressure Protection System.

Date of issuance: October 30, 2000.

Effective date: October 30, 2000.

Amendment Nos.: 208 and 202.

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

Date of initial notice in **Federal Register:** August 9, 2000 (65 FR 48751). The supplemental information provided on October 2 and 4, 2000, provided clarifying information only and did not affect the proposed no significant hazards consideration determination.

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

Brief description of amendments: The amendments clarify Technical Specification 3/4.4.4, ''Pressurizer,'' to reflect the current power supply to the pressurizer heaters and require two operable trains of pressurizer heaters during Modes 1, 2, and 3. In addition, the amendments revise the Bases for Technical Specification 3/4.4.4 to reflect these changes and clarify the purpose of the pressurizer heaters.

Date of issuance: October 20, 2000.

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

Amendment Nos.: 246 and 227.

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 56952).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 19, 2000, as supplemented on August 31, 2000.

Description of amendment request: The amendment implements a performance-based Containment Leakage Testing Program in accordance with 10 CFR Part 50, Appendix J, Option B as a substitute for the requirements of 10 CFR Part 50, Appendix J, Option A. The use of this option requires the implementation of a program based on Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' and modification of the Technical Specifications to reflect this program.

Date of issuance: November 2, 2000.

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

Amendment No.: 186.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51359).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 2000.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California

Date of application for amendment: December 31, 1999, as supplemented by letters dated January 18, July 7, September 22, and 29, and October 12, 2000.

Brief description of amendment: The amendment revises Section 2.C.(1) of Facility Operating License No. DPR–80 to authorize operation of Unit 1 at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power). Unit 2 is already authorized to operate at that power level. This amendment also revises several sections within the Improved TS to reflect the increase in reactor power level.

Date of issuance: October 26, 2000.

Effective date: October 26, 2000.

Amendment No.: Unit 1—143.

Facility Operating License No. DPR–80: The amendments revised the Technical Specifications and operating license.

Date of initial notice in ***Federal Register****:* April 19, 2000 (65 FR 21037).

The January 18, July 7, September 22, and 29, and October 12, 2000, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 27, 2000, as supplemented August 16, 2000, and September 29, 2000.

Brief description of amendment: The amendment provides for the applicability of the current safety limit minimum critical power ratio (SLMCPR), TS Section 1.1.A, to cycles beyond Cycle 14. The change also updates the approved version of the topical report in TS Section 6.9.A.4.b.1.

Date of issuance: October 30, 2000.

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

Amendment No.: 266.

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

Date of initial notice in ***Federal Register****:* August 23, 2000 (65 FR 51362).

The August 16 and September 29, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket No. 50–364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: September 8, 2000, as supplemented on October 2, 2000.

Brief Description of amendment: The amendment revises surveillance requirements 3.4.11.1 and 3.4.11.4 to eliminate the requirement to cycle the Unit 2 pressurizer power-operated relief valve block valves during the remainder of operating cycle 14.

Date of issuance: October 25, 2000.

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

Amendment No.: Unit 2—139.

Facility Operating License No. NPF–8: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 10, 2000 (65 FR 60223).

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

The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by November 9, 2000, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated October 25, 2000.

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

Date of application for amendments: November 24, 1999 (TS 99–16).

Brief description of amendments: These amendments revised the Technical Specifications (TSs) to update the industry standard that is used to test the charcoal adsorber efficiency in safety-related ventilation systems.

Date of issuance: November 2, 2000.

Effective date: November 2, 2000.

Amendment Nos.: 263 and 254.

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

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1929). The September 21, 2000, supplement provided clarifying information that did not change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 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 14, 2000, as supplemented on September 22, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to clarify the valve isolation signal information in the TS Table 4.7.2 and make an administrative change to the Table main steam isolation valves component identification.

Date of Issuance: October 31, 2000.

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

Amendment No.: 194.

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

Date of initial notice in ***Federal Register:*** September 27, 2000 (65 FR 58111).

The September 22, 2000, supplemental letter was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of November 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–29250 Filed 11–14–00; 8:45 am]

BILLING CODE 7590–01–P

PENSION BENEFIT GUARANTY CORPORATION

Interest Assumption for Determining Variable-Rate Premium; Interest Assumptions for Multiemployer Plan Valuations Following Mass Withdrawal

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of interest rates and assumptions.

SUMMARY: This notice informs the public of the interest rates and assumptions to be used under certain Pension Benefit Guaranty Corporation regulations. These rates and assumptions are published elsewhere (or are derivable from rates published elsewhere), but are collected and published in this notice for the convenience of the public. Interest rates are also published on the PBGC's web site (www.pbgc.gov).

DATES: The interest rate for determining the variable-rate premium under part 4006 applies to premium payment years beginning in November 2000. The interest assumptions for performing multiemployer plan valuations following mass withdrawal under part 4281 apply to valuation dates occurring in December 2000.

FOR FURTHER INFORMATION CONTACT: Harold J. Ashner, Assistant General Counsel, Office of the General Counsel, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005, 202–326–4024. (For TTY/TDD users, call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4024.)

SUPPLEMENTARY INFORMATION:

Variable-Rate Premiums

Section 4006(a)(3)(E)(iii)(II) of the Employee Retirement Income Security Act of 1974 (ERISA) and § 4006.4(b)(1) of the PBGC's regulation on Premium Rates (29 CFR part 4006) prescribe use of an assumed interest rate in determining a single-employer plan's variable-rate premium. The rate is the "applicable percentage" (currently 85 percent) of the annual yield on 30-year Treasury securities for the month preceding the beginning of the plan year for which premiums are being paid (the "premium payment year"). The yield figure is reported in Federal Reserve Statistical Releases G.13 and H.15.

The assumed interest rate to be used in determining variable-rate premiums for premium payment years beginning in November 2000 is 4.93 percent (*i.e.*, 85 percent of the 5.80 percent yield figure for October 2000).

The following table lists the assumed interest rates to be used in determining variable-rate premiums for premium payment years beginning between December 1999 and November 2000.

For premium payment years beginning in:	The assumed interest rate is:
December 1999	5.23
January 2000	5.40
February 2000	5.64
March 2000	5.30
April 2000	5.14
May 2000	4.97
June 2000	5.23
July 2000	5.04
August 2000	4.97
September 2000	4.86
October 2000	4.96
November 2000	4.93

Multiemployer Plan Valuations Following Mass Withdrawal

The PBGC's regulation on Duties of Plan Sponsor Following Mass Withdrawal (29 CFR part 4281) prescribes the use of interest assumptions under the PBGC's regulation on Allocation of Assets in Single-employer Plans (29 CFR part 4044). The interest assumptions applicable to valuation dates in December 2000 under part 4044 are contained in an amendment to part 4044 published elsewhere in today's **Federal Register**. Tables showing the assumptions applicable to prior periods are codified in appendix B to 29 CFR part 4044.