canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting Mr. Sam Duraiswamy (telephone 301/415–7364), between 7:30 a.m. and 4:15 p.m., EST.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m. EST at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: November 10, 1999.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 99–29992 Filed 11–16–99; 8:45 am]

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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, 1999, through November 5, 1999. The last biweekly notice was published on November 3, 1999 (64 FR 59796).

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

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

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

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

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

By December 17, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and 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 Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of amendment request: October 21, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for the Harris Nuclear Plant (HNP) to implement selected improvements described in NRC Generic Letter (GL) 93–05, "Line-Item Technical Specifications To Reduce Surveillance Requirements For Testing During Power Operation," dated September 27, 1993. Specifically, HNP proposes to modify the following TS to be consistent with GL 93–05: (1) TS 4.1.3.1.2—Change the frequency of the control rod movement test to quarterly; (2) TS 4.6.4.1—Change the frequency of the Hydrogen Monitor analog channel operational test to quarterly; (3) TS 4.3.3.1 (Table 4.3–3)—Change the Radiation Digital Channel Operational Test to quarterly; (4) TS 4.4.6.2.2.b.—Change the time for remaining in cold shutdown without leak testing the Reactor Coolant System Pressure Isolation Valves to 7 days; (5) TS 4.4.3.2—Change the testing of the capacity of pressurizer heaters to once per 18 months; (6) TS 4.6.4.2.a.—Change the Hydrogen Recombiner functional test to once per 18 months; and (7) TS 4.7.1.2.1.a—Change frequency of testing Auxiliary Feedwater Pumps to quarterly.

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

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

There are no systems being modified as a result of this change. Additionally, the way in which equipment is tested is not affected by this change. Reducing surveillance intervals for TS components (such as control rod testing) may reduce the probability of an accident (rod drop accident) by reducing actions that could cause an accident to occur (rod movement).

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

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

No system, structure, or component is being modified as a result of this change. Additionally, there are no changes to the way equipment is operated as a result of this change. Operating parameters are not being modified as a result of this change.

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

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

These proposed changes are in accordance with NRC Generic Letter 93–05, dated September 27, 1993 and NUREG–1366, dated December 1992. These changes pertain to testing requirements for TS equipment which help ensure operability requirements are met. This change does not modify the required safety function or operating parameters for equipment described in HNP TS.

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

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

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

NRC Section Chief: Kahtan Jabbour, Acting.

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

Date of amendment request: October 15, 1999.

Description of amendment request: The amendments would revise Section 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' of the Technical Specifications. Section 5.5.7 currently specifies that inspections be done according to Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, such that an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway be performed at approximately 3-year intervals. The licensee proposed to revise this to require a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one half the outer radius, or a surface examination (magnetic particle and/or penetration testing) of exposed surfaces defined by the volume of the disassembled flywheel. The licensee stated that the technical basis has been set forth in Westinghouse Topical Report WCAP–14535A, and cited similar amendments already granted to other nuclear plants.

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

First Standard

Would implementation of the changes proposed in this LAR involve a significant increase in the probability or consequences of an accident previously evaluated?

No. There are no accident probabilities or consequences impacted by this LAR [license amendment request]. As discussed in Attachment 3 [the licensee's description of the proposed amendment], following a reduction in the scope and frequency of the examinations currently required by the applicable Technical Specifications and Regulatory Guide 1.14, Revision I, an adequate inservice inspection program will continue to be maintained for the reactor coolant pump flywheels. Since the integrity of the flywheels will continue to be ensured, these components will continue to be available to fulfill their existing design function during pump coastdown flow transients. Additionally, there is no more risk that the flywheels will become a source of missile generation. Consequently, there is no significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any previously evaluated?

No. The proposed changes contained in this LAR only reduce the existing inspection requirements for the reactor coolant pump flywheels. This LAR proposes no changes to the plants' design, equipment, or method of operation at either McGuire or Catawba Nuclear Station. Furthermore, the reduction in the inspection requirements for the flywheels has been generically approved by the NRC and is justified by WCAP–14535A. Therefore, since implementation of this LAR results in no actual impact upon either of the Duke nuclear plants, and since the integrity of the flywheels will continue to be ensured at an acceptable level, no new or different kinds of accidents are being created.

Third Standard

Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. These barriers are unaffected by the changes proposed in this LAR. As discussed in WCAP–14535A, a reduction in the frequency for performing the inservice inspections currently done in accordance with Regulatory Guide 1.14, Revision I, will not preclude the ability to accurately demonstrate the integrity of the reactor coolant pump flywheels. This LAR creates no additional threat to the integrity of the fission product barriers from the standpoint of missile generation or otherwise. Therefore, implementation of the changes proposed in this LAR does not impact the assumption of the integrity of the flywheels, the fission product barriers, or any other accident analyses assumptions. Consequently, no margin of safety will be significantly impacted by this LAR.

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: October 15, 1999.

Description of amendment request: The amendments would revise Section 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' of the Technical Specifications. Section 5.5.7 currently specifies that inspections be done according to Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, such that an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway be performed at approximately 3-year intervals. The licensee proposed to revise this to require a qualified in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one half the outer radius, or a surface examination (magnetic particle and/or penetration testing) of exposed surfaces defined by the volume of the disassembled flywheel. The licensee stated that the technical basis has been set forth in Westinghouse Topical Report WCAP–14535A, and cited similar amendments already granted to other nuclear plants.

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

First Standard

Would implementation of the changes proposed in this LAR involve a significant increase in the probability or consequences of an accident previously evaluated?

No. There are no accident probabilities or consequences impacted by this LAR [license amendment request]. As discussed in Attachment 3 [the licensee's description of the proposed amendment], following a reduction in the scope and frequency of the examinations currently required by the applicable Technical Specifications and Regulatory Guide 1.14, Revision I, an adequate inservice inspection program will continue to be maintained for the reactor coolant pump flywheels. Since the integrity of the flywheels will continue to be ensured, these components will continue to be available to fulfill their existing design function during pump coastdown flow transients. Additionally, there is no more risk that the flywheels will become a source of missile generation. Consequently, there is no significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any previously evaluated?

No. The proposed changes contained in this LAR only reduce the existing inspection requirements for the reactor coolant pump flywheels. This LAR proposes no changes to the plants' design, equipment, or method of operation at either McGuire or Catawba Nuclear Station. Furthermore, the reduction in the inspection requirements for the flywheels has been generically approved by the NRC and is justified by WCAP–14535A. Therefore, since implementation of this LAR results in no actual impact upon either of the Duke nuclear plants, and since the integrity of the flywheels will continue to be ensured at an acceptable level, no new or different kinds of accidents are being created.

Third Standard

Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. These barriers are unaffected by the changes proposed in this LAR. As discussed in WCAP–14535A, a reduction in the frequency for performing the inservice inspections currently done in accordance with Regulatory Guide 1.14, Revision I, will not preclude the ability to accurately demonstrate the integrity of the reactor coolant pump flywheels. This LAR creates no additional threat to the integrity of the fission product barriers from the standpoint of missile generation or otherwise. Therefore, implementation of the changes proposed in this LAR does not impact the assumption of the integrity of the flywheels, the fission product barriers, or any other accident analyses assumptions. Consequently, no margin of safety will be significantly impacted by this LAR.

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

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: September 29, 1999.

Description of amendment request: The proposed amendments would revise the Containment Inservice Inspection (ISI) Program Technical Specifications (TS) 5.5.2, ''Containment Leakage Testing Program,'' and TS 5.5.7, ''Pre-Stressed Concrete Containment Tendon Surveillance Program.'' The proposed amendments would permit the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsection IWL visual examinations to be performed in lieu of concrete and post-tensioning system general visual examinations required by 10 CFR 50, Appendix J and Regulatory Guide 1.163 between Type A tests. In addition, the amendment would permit general visual examinations of the concrete and post-tensioning system that can be performed with a unit in operation to be performed prior to the beginning of a refueling outage during which a Type A test is scheduled.

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

No. Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no significant effect on accident probabilities or consequences. The containment is not an accident initiating system or structure; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The containment serves an important function to mitigate consequences of postulated accidents previously evaluated and the examination frequencies proposed in this amendment will not result in a reduction in the capacity of the containment to meet its intended function. The requested flexibility in scheduling containment visual examinations has no significant impact on the validity of the examinations or of containment structural integrity.

Additionally, the change to Technical Specification 5.5.7 and the planned revision to Selected Licensee Commitment 16.6.2 described in this amendment application reflect the adoption of an ASME Section XI, Subsection IWE and IWL Inservice Inspection Program as required by 10 CFR 50 Section 55a(g)(4). Implementation of this program will not result in a reduction in the capacity of the containment to meet its intended function.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by approval of the requested changes.

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

No. Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant that would introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators, since the containment functions primarily as an accident mitigator.

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

No. Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation, including the performance of the containment. This component is already capable of performing as intended, and its function is verified by visual examination, post-tensioning system examinations, and leakage rate testing.

The examination requirements of ASME XI, Subsection IWL, are essentially identical to those contained in Regulatory Guide 1.35, Rev. 3, and are more rigorous than those required by 10 CFR 50, Appendix J and Regulatory Guide 1.163. Previous visual examinations of containment concrete and post-tensioning system surfaces have not revealed any indications of abnormal degradation of the containment. The five-year frequency for IWL examinations is adequate in lieu of the general visual examination frequency specified in Regulatory Guide 1.163 for containment concrete and post-tensioning system examinations.

The ability of the containment to perform its design function will not be impaired by the implementation of this amendment at Oconee Nuclear Station. Consequently, no safety margins will be impacted.

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: Anne W. Cottingham, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: June 17, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.4.9.1 and associated figures to extend the applicability of the heatup and cooldown curve pressure and temperature limits from 10 effective full power years (EFPY) to 15 EFPY. The proposed changes include new heatup and cooldown curves developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2, and Code Case N–640. The applicability of TS Section 3.4.9.3, Overpressure Protection Systems, is also updated to 15 EFPY, and the maximum allowable power operated relief valve (PORV) setpoints for the over pressure protection system are revised. Revisions to the TS Bases are also made.

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

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

The proposed heatup and cooldown curves have been revised by changing the applicability from 10 effective full power years (EFPY) to 15 EFPY. The curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2 and Code Case N–640. The proposed heatup and cooldown curves define limits that still ensure the prevention of nonductile failure for the reactor vessel. The design basis events that were protected against have not changed; therefore, the probability of an accident is not increased.

The overpressure protection system (OPPS) has been revised such that the applicability has changed from 10 EFPY to 15 EFPY. This system protects the Reactor Coolant System (RCS) at low temperatures so that the integrity of the Reactor Coolant Pressure Boundary (RCPB) is not compromised by violating the pressure/temperature (P/T) limits. These changes were determined in accordance with the methodologies set forth in the regulations to provide an adequate margin of safety to ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events. The lower limit on pressure during the design basis OPPS mass injection and heat addition transients is established based on operational consideration for the RCP number one seal limit which requires a nominal differential pressure across the seal faces for proper film-riding performance. As part of the OPPS setpoint evaluation, margin to the RCP number one seal limit is evaluated.

This limit corresponds to a differential pressure across the seal of 200 psid, which corresponds to the gage pressures. The pressure undershoot below the PORV setpoint during a design basis mass injection or heat addition event can exceed 100 psi. Therefore, with the PORV setpoints developed for the 15 EFPY heatup and cooldown curves, there is the potential for RCS pressure to violate the RCP number one seal limit at the lowest RCS temperatures.

Undershoot below the PORV setpoint can be significantly higher if both PORVs actuate during an OPPS event, and it is anticipated that the pump seal limit would be exceeded. However, staggering the setpoints minimizes the likelihood that both PORVs will actuate simultaneously during credible OPPS events. Similarly, WCAP 14040–NP–A indicates that when there is insufficient range between the upper and lower pressure limits to select PORV setpoints that provide protection against violating both limits, then the setpoint selection that provides protection against the upper limit violation takes precedence. WCAP–4040–NP, Revision 1 was approved by the NRC by letter dated October 16, 1995, which was incorporated in Revision 2 of the approved WCAP issued in January 1996.

Modification of the heatup and cooldown curves and OPPS setpoints does not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the Updated Final Safety Analysis Report (UFSAR). Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed heatup and cooldown curves applicable for the first 15 EFPY were generated using approved methodology and Code Case N–640. Generating these curves with Code Case N–640 reduced the excess conservatism that exists in the current curves and results in an increase in the safety of the plant, as the likelihood of RCP seal failures and/or fuel problems will decrease. The change does not cause the initiation of any accident nor create any new single failure.

The modification of the OPPS setpoints ensures that the RCPB integrity is protected at low temperatures. The new setpoints were selected using conservative assumptions to ensure that sufficient margin is available to prevent violation of the P/T limits due to anticipated mass and heat input transients. The modification of the setpoints does not change, degrade, or prevent the safe response of the RCS to accident scenarios, as described in UFSAR Chapter 15. The proposed change does not cause the initiation of any accident nor create any new credible single failure.

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

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

The new P/T curves define the limits for ensuring prevention of nonductile failure for the reactor vessel, and does not significantly reduce the margin of safety for the plant. The methodology provided in Code Case N–640 removed some of the excess conservatism from the current Appendix G analysis. However, this improved overall plant safety by expanding the operating window relative to the RCP seal requirements. The probability of damaging the RCP seals is reduced. Therefore, the margin of safety is not significantly reduced.

The OPPS setpoints will continue to ensure the RCS pressure boundary will be protected from pressure transients. They were generated using the proposed heatup and cooldown curves as input. The OPPS setpoints include additional margin by including instrument uncertainties not included in the current setpoints. Therefore, the margin of safety is not significantly reduced.

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 15, 1999.

Description of amendment request: The license amendment request (LAR) proposes to revise the Technical Specifications frequency for the Quench and Recirculation Spray Systems nozzle air flow test from 5 years to 10 years. This LAR also includes a revision to correct the terminology used in an action requirement as well as miscellaneous editorial and format 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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed extension of the testing frequency of the Quench Spray and Recirculation Spray Systems' nozzles to ten years does not change the way these systems are operated or their operability requirements. The proposed change to the surveillance frequency of safety equipment has no impact on the probability of an accident occurrence nor can it create a new or different type of accident. NUREG–1366, "Improvements to Technical Specifications Surveillance Requirements," dated December 1992, and Generic Letter 93–05, "Line Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," dated September 27, 1993, concluded that the corrosion of stainless steel piping is negligible during the extended surveillance interval for nozzle testing. The results of the above NRC study were evaluated by Duquesne Light Company and found to be applicable to Beaver Valley Power Station (BVPS) Unit 1 and 2. Since the Quench Spray and Recirculation Spray Systems are maintained dry, there is no additional mechanism that could cause blockage of the spray nozzles. Thus, the nozzles in these spray systems are expected to remain operable during the ten year surveillance interval to mitigate the consequence of an accident previously evaluated. No obstructed or clogged spray systems' nozzles have been observed during the five year frequency surveillance tests at either BVPS Unit 1 or Unit 2 to date. Testing of the spray systems' nozzles at the proposed reduced frequency will not increase the probability of occurrence of a postulated accident or the consequences of an accident previously evaluated.

This license amendment also revises the Action criteria in the BVPS Unit 1 and 2 Axial Flux Difference [AFD] technical specification to correct the terminology referring to the Core Operating Limits Report (COLR) limits. The proposed change incorporates the terminology (acceptable operation limits) used in the corresponding Action condition of the ISTS [Improved Standard Technical Specifications]. The proposed change does not alter the AFD limits specified in the COLR and the AFD specification continues to assure plant operation within those limits. With AFD within the acceptable operation limits specified in the COLR, the resulting axial power distribution remains within the initial conditions assumed in the safety analyses. Therefore, these changes will not increase the probability of occurrence of a postulated accident or the consequences of an accident previously evaluated.

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

The proposed reduced frequency testing of the Quench Spray and Recirculation Spray Systems' nozzles does not change the way the spray systems are operated. The reduced frequency of testing the spray nozzles does not change the plant operation or system readiness. The reduced frequency testing of the Quench Spray and Recirculation Spray Systems' nozzles does not generate any new accident precursors. Therefore, the possibility of a new or different kind of accident previously evaluated is not created by the proposed changes in surveillance frequency of the spray systems' nozzles.

This license amendment also revises the Action criteria in the BVPS Unit 1 and 2 Axial Flux Difference technical specification to correct the terminology referring to the Core Operating Limits Report (COLR) limits. This addresses an incorrect use of terminology and the revision does not involve a technical intent change. Therefore, the possibility of a new or different kind of accident previously evaluated is not created by the proposed terminology correction.

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time.

Reduced testing of the Quench Spray and Recirculation Spray Systems' nozzles does not change the way these spray systems are operated or these spray systems' operability requirements. Generic Letter 93–05 and NUREG–1366 concluded that the corrosion of stainless steel piping is negligible during the extended surveillance interval for nozzle testing. The results of the above NRC study were evaluated by Duquesne Light Company and found to be applicable to BVPS Unit 1 and 2. Since the Quench Spray and Recirculation Spray Systems are maintained dry, there is no additional mechanism that could cause blockage of these spray systems' nozzles. Thus, the proposed reduced testing frequency is adequate to ensure spray nozzle operability. The surveillance requirements do not affect the margin of safety in that the operability requirements of the Quench Spray and Recirculation Spray Systems remain unaltered. The existing safety analyses remain bounding. Therefore, the margin of safety is not adversely affected.

This license amendment also revises the Action criteria in the BVPS Unit 1 and 2 Axial Flux Difference technical specification to correct the terminology referring to the Core Operating Limits Report (COLR) limits. This addresses an incorrect use of terminology and the revision does not involve a technical intent change. The operating criteria on Axial Flux Difference are not altered from their intended requirements. Therefore, the margin of safety is not adversely affected by the proposed terminology correction.

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: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Sheri R. Peterson

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

Date of amendment request: July 20, 1999

Description of amendment request: The licensee amendment request proposes to relocate the following Technical Specifications items to the Licensing Requirements Manual:

In-core Detectors (Unit 1 and 2),
Chlorine Detection System (Unit 1 and 2),
Turbine Over-speed Protection (Unit 2 only),
Crane Travel Spent Fuel Storage Pool Building (Unit 1 and 2).

In addition to the relocation, certain editorial and format changes are proposed. Also, it is proposed that certain information on the Remote Shutdown Panel Monitoring Instrumentation be moved to the Updated Final Safety Analysis Report (USFAR).

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

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

Consistent with the guidance provided in Generic Letter (GL) 95–10 and the content of the Improved Standard Technical Specifications (ISTS) contained in NUREG–1431, Rev. 1, this license amendment request (LAR) proposes the relocation of the following TS to the Licensing Requirements Manual (LRM):

3/4.3.3.2 Incore Detectors (Unit 1 and 2)
3/4.3.3.7 Chlorine Detection System (Unit 1 and 2)
3/4.3.4 Turbine Overspeed Protection (Unit 2 only)

In order to completely relocate the chlorine detection system requirements from the Technical Specifications (TS), portions of the Unit 1 Specifications 3/4.7.7, Control Room Habitability Systems and 3/4.9.15, Control Room Emergency Habitability Systems, as well as the Unit 2 Specification, 3/4.7.7, Control Room Emergency Air Cleanup and Pressurization System are proposed to be revised to reflect the removal of the chlorine detection system from the TS. The applicable surveillance requirements, and modes of applicability from these specifications are proposed to be relocated to the LRM along with the associated chlorine detection system TS. In addition, new actions have been added to the chlorine detection system specifications to integrate the new requirements.

In addition to the TS identified for relocation by the NRC in GL 95–10, this LAR proposes the relocation of another TS that does not meet the criteria of 10 CFR 50.36 and is not included in the ISTS. The additional TS proposed to be relocated to the LRM is 3/4.9.7 Crane Travel Spent Fuel Storage Pool Building (Unit 1 and 2).

This LAR also proposes that the TS Bases section associated with each of the TS listed above be relocated to the LRM as well. The appropriate TS pages (i.e., LCO, Bases, Table of Contents, etc.) are revised to reflect the removal of these Specifications and Bases from the TS.

The TS and bases discussed above and proposed for relocation will be moved into the BVPS LRM. The Unit 1 and Unit 2 LRM are appendices of the associated unit UFSAR. As part of the UFSAR any changes made to the LRM must be in accordance with the provisions of 10 CFR 50.59.

In addition to the relocation of the above listed TS, this LAR includes the removal of the "Measurement Range" information from the Unit 1 and 2 TS Table 3.3–9, Remote Shutdown Panel Monitoring Instrumentation. This design information is being moved from the TS to an applicable Updated Final Safety Analysis Report (UFSAR) section. The removal of this detail from the TS is consistent with the level of detail in the corresponding ISTS Specification. As part of the UFSAR any changes made to the measurement range information must be in accordance with the provisions of 10 CFR 50.59.

LAR 1A–251/2A–121 includes two Bases enhancements. Additional information is being added to the reactor trip system instrumentation Bases to discuss diverse and anticipatory protection features not credited in the accident analyses. The reactor trip system instrumentation Bases is also revised to more clearly describe the source and intermediate range neutron flux protection features required during shutdown modes.

The proposed changes include the addition of license numbers to some of the TS pages contained in this LAR. In addition, this LAR contains changes that update the format of the affected TS pages and make editorial corrections. These changes are administrative in nature and do not impact the technical content of the affected TS pages.

The proposed changes regarding the relocation of information from the TS in this LAR follow the guidance provided in Generic Letter 95–10, the NRC ''Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors'' (58 FR 39132) dated July 22, 1993, and are consistent with the content of the ISTS. In addition, the proposed location for this information (UFSAR and LRM) ensures that future changes to the relocated requirements will be in accordance with the provisions of 10 CFR 50.59 and that NRC review and approval will be requested should a change to this information involve an unreviewed safety question.

The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any accident initiator. No analyzed accident scenario is being changed. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed. The failure of any of the systems or components affected by this LAR, except for turbine overspeed protection, is not an accident initiating event. Due to the low likelihood of equipment damage or failure resulting from turbine missiles generated by a turbine overspeed event, assumptions related to the turbine overspeed protection system are not part of an initial condition of a design basis accident or transient.

The proposed amendment also does not involve a significant increase in the consequences of an accident previously evaluated. The amendment does not reduce the current requirements for the systems and components proposed for relocation. The amendment only requests that the requirements be retained in a more appropriate document. The systems and components proposed for relocation in this amendment perform no active role in mitigating a design basis accident described in the UFSAR. The systems or components proposed for relocation are not part of the initial conditions assumed in a safety analysis for a design basis accident described in the UFSAR. In addition, the affected systems and components do not function to actuate any protective equipment, nor are they part of the primary success path assumed in the safety analyses to mitigate any design basis accident described in the UFSAR.

The bases enhancements included in this LAR are administrative in nature and serve only to provide additional descriptive information. These changes do not impact plant safety.

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

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

The proposed amendment does not involve any physical changes to the plant or the modes of plant operation defined in Appendix A of the operating license. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems.

Moving specifications to the LRM or design information to the UFSAR will not change the physical plant or the modes of plant operation. Whether these specifications are located in the TS or the LRM has no effect on any previously evaluated accident. The relocation of TS information does not involve a change in the configuration of equipment nor does it alter the design or operation of plant systems.

Expanding the Bases for both units to discuss additional information regarding the protective functions not credited in the safety analysis or the neutron flux trip functions required in shutdown modes provides additional information to enhance the awareness of the protective instrumentation functions. The proposed bases changes do not result in any adjustments or physical alteration to the affected protective instrumentation functions. The Reactor Protection System will continue to function as currently designed and assumed in the accident analyses.

Therefore, operation of the facility 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. Does the change involve a significant reduction in a margin of safety?

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time, and sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The relocation of TS requirements and information to the LRM or UFSAR does not reduce the requirements for the affected systems and components to be maintained operable and function within design requirements. The relocation of TS requirements and information to the LRM and UFSAR will allow changes to this information to be made in accordance with the provisions of 10 CFR 50.59 and continues to ensure that NRC review and approval will be requested should a change to this information involve an unreviewed safety question.

Expanding the Bases for both units to discuss additional information regarding the protective functions not credited in the safety analysis or the neutron flux trip functions required in shutdown modes provides additional information to enhance the awareness of the protective instrumentation functions. The addition of descriptive text to the TS bases does not affect the TS requirements for the affected equipment to be maintained operable and function within the applicable design requirements. The Reactor Protection System will continue to function as currently designed and assumed in the accident analyses.

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

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: September 20, 1999.

Description of amendment request: The proposed amendments would revise the standard to which the control room ventilation charcoal and Supplementary Leak Collection and Release System (SLCRS) charcoal must be laboratory tested as specified in: Beaver Valley Power Station, Unit No. 1 (BVPS–1), Technical Specification (TS) 4.7.7.1.1.c.2 for the Control Room Emergency Habitability Systems; BVPS–1 TS 4.7.8.1.b.3 for the SLCRS; Beaver Valley Power Station, Unit No. 2 (BVPS–2), TS 4.7.7.1.d for the Control Room Emergency Air Cleanup and Pressurization System; and BVPS–2 TS 4.7.8.1.b.3 for the SLCRS. NRC Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999, requested licensees to revise their TS criteria associated with laboratory testing of ventilation charcoal to a valid test protocol, which included American Society for Testing Materials (ASTM) D3803–1989. This license amendment request revises the charcoal laboratory standard to follow ASTM D3803–1989 for each BVPS Unit.

This license amendment request also: (1) Revises the minimum amount of output in kilowatts needed for the control room emergency ventilation system heaters at each BVPS Unit; (2)

revises BVPS–1 SLCRS surveillance testing criteria to be consistent with American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) N510–1980, the BVPS–1 control room ventilation testing, and the BVPS–2 SLCRS/control room ventilation testing; and (3) makes minor typographical corrections and editorial 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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes to the surveillance requirements for the laboratory testing of ventilation system charcoal are consistent with Generic Letter 99–02. The proposed change will adopt ASTM D3803–1989 as the laboratory testing standard for performing the surveillance associated with the Control Room emergency ventilation and the SLCRS charcoal filters at each BVPS Unit. Thus this proposed change will not involve a significant increase in the probability or consequences of a previously evaluated accident since this standard provides the assurance for continuing to comply with the current BVPS Unit 1 and Unit 2 licensing basis as it relates to the dose limits of GDC 19 and 10 CFR Part 100.

The change in the control room emergency ventilation system heater minimum output at both BVPS Units does not change the system ability to meet its design bases. The change in the BVPS Unit 1 SLCRS testing frequency for adsorber/filter in-place testing and the adsorber laboratory testing does not change the SLCRS system's ability to meet its design bases. The change in the BVPS Unit 1 SLCRS testing frequency for SLCRS air flow distribution testing does not change the SLCRS system's ability to meet its design bases. Therefore, these changes will not increase the probability of occurrence of a postulated accident or the consequences of an accident previously evaluated since these systems' ability to operate as required remains unchanged.

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

The proposed license amendment to the control room emergency ventilation system and SLCRS at both BVPS Units does not change the way the system is operated. The proposed changes only involve changes to the surveillance testing. These testing modifications do not alter these systems' ability to perform their design bases. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated accident since the control room emergency ventilation system and SLCRS will continue to operate in accordance with their previous design bases.

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not affect the ability of system, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time.

The proposed license amendment to the control room emergency ventilation system and SLCRS at both BVPS Units does not change the way the system is operated. The proposed changes only involve changes to the surveillance testing. These testing modifications do not alter these systems' ability to perform their design bases. The existing safety analyses remain bounding. Therefore, the margin of safety is not adversely affected.

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: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: September 22, 1999.

Description of amendment request: The proposed amendment would allow a one-time only extension to the surveillance interval of Technical Specification Surveillance 4.7.12.d for functional testing of snubbers. The proposed extension would be limited to the end of the 8th refueling outage or November 30, 2000, whichever occurs sooner.

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

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

The proposed change is for a one-time extension to the surveillance interval for functional testing of snubbers specified in Technical Specification (TS) 4.7.12.d. The proposed change involves revising the calendar time allowed between functional tests and would result in a maximum surveillance interval extension of approximately 6.5 months.

The proposed change continues to adequately limit plant operation between required snubber surveillances by ensuring the required surveillances are performed by November 30, 2000. Therefore, the proposed change continues to limit snubber wear due to vibration and elevated temperatures. The elevated temperatures and vibration experienced during plant operation are the primary contributors to snubber wear.

In addition, snubber-testing experience has shown that the historical failure rate of snubbers is low. There have been seven refueling outages since Unit 2's startup in 1987. Only during the first refueling outage, 2R01, did the snubber functional test sample plan identify any inoperable snubbers. In that outage, seven snubbers tested inoperable. All failed due to damage sustained during original construction and startup activities. Since 2R01, no inoperable snubbers were found by sample plan functional testing performed during each surveillance interval. Also, the latest visual inspections performed on the Unit 2 snubbers (during 2R07) revealed no evidence of damage or potential problems with any snubber.

Due to the low incidence of snubber functional test failures resulting from sample plan testing and the limited plant operating time between tests, the possibility of a snubber failure resulting from this one-time surveillance extension is low. No changes are being made to any accident initiator. No analyzed accident scenario is being changed. The initiating conditions and assumptions of previously analyzed accidents remain unchanged. Therefore, the proposed change does not involve a significant increase in the probability of a previously evaluated accident.

This change does not involve a physical change to the plant and does not affect the acceptance criteria specified in the TS for snubber functional testing, nor does this change reduce the remedial actions required for inoperable snubbers. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed amendment does not involve any physical changes to the plant or the modes of plant operation defined in Appendix A of the operating license. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. The one-time surveillance interval extension proposed by this change will not reduce the capability of the snubbers to perform their design function.

Therefore, operation of the facility 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. Does the change involve a significant reduction in a margin of safety?

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

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time, and sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The proposed change is for a one-time extension to the surveillance interval for functional testing of snubbers specified in Technical Specification 4.7.12.d. The proposed change continues to adequately limit plant operation between required snubber surveillances by ensuring the required surveillances are performed by November 30, 2000. Therefore, the proposed change continues to limit snubber wear due to vibration and elevated temperatures. The elevated temperatures and vibration experienced during plant operation are the primary contributors to snubber wear.

In addition, snubber-testing experience has shown that the historical failure rate of snubbers is low. There have been seven refueling outages since Unit 2's startup in 1987. Only during the first refueling outage, 2R01, did the snubber functional test sample plan identify any inoperable snubbers. In that outage, seven snubbers tested inoperable. All failed due to damage sustained during original construction and startup activities. Since 2R01, no inoperable snubbers were found by sample plan functional testing performed during each surveillance interval. Also, the latest visual inspections performed on the Unit 2 snubbers (during 2R07) revealed no evidence of damage or potential problems with any snubber.

This change does not involve a physical change to the plant and does not affect the acceptance criteria specified in the TS for snubber functional testing, nor does this change reduce the remedial actions required for inoperable snubbers. The snubbers and systems supported by the snubbers will continue to be available to perform their intended safety functions during the requested extension period.

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

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: July 30, 1999.

Description of amendment request: The request proposes changes to the Technical Specifications (TSs) and the operating license to extend operation of the station from its licensed power of 2894 megawatts thermal (MWt) to the uprated power level of 3039 MWt, an increase of 5 percent. The proposed changes are to (1) extend the definition of rated thermal power in TS Section 1.1 and the operating license to 3039 MWt; (2) reduce the thermal power safety limit of TSs 1.4, 2.1.1.1, 3.2.1, 3.2.2, 3.2.3, 3.3.1.1, 3.4.3, and 3.7.5; (3) increase the reactor steam dome pressure in TS Table 3.1.4–1, TS 3.4.12, and SR 3.5.3.3; (4) increase the control rod drive charging water header pressure in TSs 3.1.5, 3.9.5, and 3.10.8; (5) increase the standby liquid control (SLC) system Boron-10 enrichment and concentration criteria in TS 3.1.7; (6) increase the surveillance test discharge pressure for the SLC pump in surveillance requirement (SR) 3.1.7.7; (7) increase the allowable value of the reactor vessel steam dome pressure—high scram setpoint in TS Table 3.3.1.1–1; (8) increase the allowable value for the anticipated transient without scram—reactor pressure trip reactor steam dome pressure—high setpoint in SR 3.3.4.2.4; (9) revise the safety, relief, and low low set function of the main steam safety/relief valves (SRVs) in SRs 3.3.6.4.3 and 3.4.4.1; (10) increase the upper and lower bounds on reactor pressure for the purposes of performing reactor core isolation cooling pump flow rate surveillance at high pressure in SR 3.5.3.3; (11) increase the main steam line flow—high reactor isolation trip in TS Table 3.3.6.1–1; (12) reduce the thermal power limits for single loop operation in TS 3.4.1; (13) increase the upper and lower bounds on reactor pressure for the purposes of performing pressure isolation valve surveillance at high pressure in SR 3.4.6.1; and (14) revise the reactor coolant system pressure/temperature limits in TS 3.4.11 (including replacing TS Figure 3.4.11–1 with figures for 14 and 32 effective full power years of operation). Item (9) includes increasing the main steam SRV setpoint tolerance from +0%, −2% to [plus or minus] 3% in SR 3.4.4.1.

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

(1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level, as the regulatory criteria established for plant equipment (ASME [American Society of Mechanical Engineers] Code, IEEE [Institute of Electrical and Electronic Engineers] standards, NEMA [National Equipment Manufacturers Association] standards, Reg[ulatory] Guide criteria, etc.) will still be complied with at the uprated power level. An evaluation of the BWR [boiling water reactor] probabilistic risk assessments concludes that the calculated core damage frequencies will not significantly change due to [the] power uprate. Scram setpoints (equipment settings that initiate automatic plant shutdowns) will be established such that there is no significant increase in scram frequency due to [the] uprate. No new challenges to safety-related equipment will result from [the] power uprate.

The changes in consequences of hypothetical accidents which would occur from 102% of the uprated power, compared to those previously evaluated from [greater than or equal to] 102% of the original power, are in all cases insignificant, because the accident evaluations from [the] power uprate to 105% of original power ([approximately] 106% of original steam) flow will not result in exceeding the NRC-approved acceptance [criteria] limits. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met at the uprated power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO–24011, ''GESTAR II''. Challenges to fuel or ECCS [emergency core cooling system] performance are evaluated, and shown to still meet the criteria of 10 CFR 50.46 and Appendix K [to 10 CFR 50], (Section 4.3 above, and Regulatory Guide 1.70 and USAR [Updated Safety Analysis Report] Section 6.3).

Challenges to the containment have been evaluated, and the containment and its associated cooling systems will continue to meet 10 CFR 50 Appendix A [General Design Criteria] Criterion 38, Long Term Cooling, and Criterion 50, Containment.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 100 (Regulatory Guide 1.70 & USAR Chapter 15).

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

As summarized below, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by [the] power uprate has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations defined in Regulatory Guide 1.70 have been evaluated and no new or different kind of accident has been identified. [The power] Uprate uses already developed technology, and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria to include NRC approved codes, standards, and methods. GE [General Electric] has designed BWRs of higher power levels than the uprated power of any of the currently operating BWR fleet and no new power dependent accidents have been identified.

The Technical Specification changes needed to implement [the] power uprate require some small adjustments, but no change to the plant's physical configuration. All changes have been evaluated, and are acceptable.

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

The calculated loads on all affected structures, systems and components will remain within their design allowables for all design basis event categories. No NRC acceptance criteria will be exceeded. Only some design and operational margins are affected by [the] power uprate. The margins of safety originally designed into the plant are not affected by [the] power uprate. Because the plant configuration and reactions to transients and hypothetical accidents will not result in exceeding the presently approved NRC acceptance limits, [the] power uprate can 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 for the power uprate.

Although not required for the power uprate, the licensee also requested a change to technical specifications to increase the main steam SRV setpoint tolerance from +0%, −2% to [plus or minus] 3%. However, the licensee's no significant hazards consideration for the power uprate does not expressly address the change to the SRV setpoint tolerance. Therefore, the NRC staff's review of this change is presented below:

(1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The main steam SRV's safety function lift setpoints are tested in accordance with ASME Code requirements and the licensee's inservice testing program. The setpoint tolerance determines whether the SRV passes or fails the surveillance requirement and if additional valves are to be tested. Notwithstanding the results of the safety function lift setpoint test, if the measured value is outside a tolerance of [plus or minus] 1%, the valve is reset to within [plus or minus] 1% of the design lift setpoint. Therefore, the change to the SRV setpoint tolerance does not affect the performance of any structure, system, or component in the plant and does not affect the operation of the plant. Accordingly, the change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The setpoint tolerance change does not alter the function of the valves' over-pressure protection features, and the release of steam/water through the SRVs is addressed in previously evaluated accident analysis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The change only affects whether a SRV passes or fails its safety function surveillance requirement, as well as the total number of valves to be tested. Regardless the outcome of these tests, all valves tested will be returned to within [plus or minus] 1% of the design lift setpoint. The 2% nominal "as-left" tolerance span is effectively the same tolerance span as specified in the current technical specifications. As a result, there is no significant reduction in a margin of safety.

Therefore, based on its review, it appears that the three standards of 10 CFR 50.92(c) are satisfied, and the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 3, 1999.

Description of amendment request: Entergy Operations, Inc. (licensee) has proposed to revise Final Safety Analysis Report (FSAR) Section 9.5.4.1, "Diesel Generator Fuel Oil Storage and Transfer Systems." The revision will change this section of the FSAR to explicitly list the Waterford Steam Electric Station, Unit 3 (Waterford 3) deviations from the guidance described in American National Standards Institute (ANSI) N195–1976, "Fuel Oil Storage System for Standby Diesel Generator." The licensee determined that these proposed changes require Nuclear Regulatory Commission staff approval prior to implementation.

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

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

Response: No.

The proposed change revises the Waterford 3 FSAR to match the current design of the Waterford 3 fuel oil storage and transfer system. The change effectively requests deviations from portions of ANSI N195–1976. None of these changes significantly increases the probability of an accident because the Emergency Diesel Generator (EDG) fuel oil system is not an initiator of any analyzed event. There are no accidents analyzed in the Final Safety Analysis Report (FSAR) that are initiated by the systems or components affected by these changes.

The deviation from ANSI N195–1976, which allows less than the ANSI Standard recommended volume to be stored in the existing EDG Fuel Oil Storage Tanks (FOSTs) A and B, will not significantly increase the consequences of an accident. Waterford 3 contains at least seven days of fuel oil in each FOST. Although the Waterford 3 FOSTs do not contain a 10% margin, there are numerous diesel fuel oil vendors nearby from which to obtain fuel oil. Waterford 3 also has the capability to transport EDG fuel oil from vendors by tanker truck, train, or barge. This situation ensures that Waterford 3 will have fuel oil readily available when there is a need for replenishment. Waterford 3 does not store the additional amount of fuel oil required for testing. A previous Technical Specification (TS) Amendment addressed the Waterford 3 FOSTs not containing enough fuel oil for testing. However, an exception to this requirement was previously approved in TS Amendment 92.

The request for deviation from the ANSI N195–1976 requirement for the feed tank suction to be from above the bottom, will not increase the consequences of any accident. Previous operating experience at Waterford 3 has shown that since initial startup there have not been any water or filter blockage problems attributed to the bottom suction from the feed tank. The fuel oil in each feed tank is replenished every 31 days during the EDG monthly Surveillance Requirement (SR). Blockage problems are further minimized because testing the FOSTs for particulates is performed with a more conservative filter size than installed on the EDG engine (0.8

microns versus 5 microns). Also, TS Surveillances require water and sediment content to be verified and if water is present, for it to be removed.

The request for deviation from the ANSI N195–1976 requirement for the feed tank overflow to discharge to the FOST will not increase the consequences of any accident. The feed tank is equipped with design features to ensure fuel oil is not depleted due to over-filling the feed tank. The feed tank contains a high level switch that stops the transfer pump upon indication of high level and a high level alarm that alerts the Control Room of high level in the tank. A failure of both the feed tank high level switch and high level alarm occurring simultaneously is very remote. These measures will not prevent the loss of some fuel oil; however, two failures would have to occur to prevent the Control Room from being notified. Even if one EDG FOST were depleted because of the above failures, the other EDG FOST would be available to ensure seven days of fuel oil for one EDG.

The request for deviation from the ANSI N195–1976 requirement to have one pressure indicator located in the discharge of the fuel oil transfer pump will not increase the consequences of any accident. A pressure indicator on the discharge of the transfer pump could indicate performance degradation of the pump; however, the Waterford 3 transfer pumps are designed for automatic operation. If a failure of the transfer pump occurred, indication would appear in the Control Room via the alarm for low feed tank level. The alarm for low feed tank level is adequate to alert the Control Room of a transfer pump malfunction. If a transfer pump were to malfunction, the other transfer pump would be available to deliver fuel oil to operate one EDG for at least seven days. ASME Section XI testing is performed on the transfer pump once per quarter (temporary pressure instrumentation is installed on the discharge of the pump to measure pump differential pressure) to verify that pump performance has not degraded. In addition, the transfer pumps are functionally tested every month during routine testing of the EDGs.

The requested deviations from ANSI N195–1976 do not affect the consequences of an accident because none of the requested deviations will prevent the EDG from having seven days of fuel oil available (without multiple failures). Therefore, the EDG fuel oil system will perform as required to provide sufficient fuel oil to the EDG to mitigate the consequences of design basis accidents.

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

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

Response: No.

The proposed change revises the Waterford 3 FSAR to match the current design of the Waterford 3 fuel oil storage and transfer system. This change is a change to a commitment, and has no [a]ffect on the current diesel fuel oil storage system or how it is operated, nor does it [a]ffect any other safety systems or components, or the way the plant is operated. The change does not affect any accident analysis assumptions (including a loss of offsite power) or accident analysis conclusions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change revises the Waterford 3 FSAR to match the current design of the Waterford 3 fuel oil storage system. Although Waterford 3 deviates from certain ANSI N195–1976 requirements, these deviations do not result in any changes to the fuel oil storage system or accident analyses. The deviations do not affect the ability of any safety systems required to protect the multiple barriers. No accident mitigatiors are affected by the change because the amount of available fuel oil has not changed. As a result, the proposed deviations will not cause a significant decrease in the margin of safety or prevent Waterford 3 from safely shutting down. The result of using Probabilistic Safety Assessment techniques conclude that increasing the fuel oil storage capacity at Waterford 3 to comply with the ANSI requirements has no risk significance. The specific [a]ffects of the deviations on the margin of safety are addressed below.

The current TS for stored EDG fuel oil ensures there is sufficient fuel oil to operate one EDG for seven days assuming the worst case single active or passive failure. Fuel oil is readily available due to the number of vendors in the vicinity of Waterford 3. Waterford 3 is also capable of replenishing EDG fuel oil via tanker truck, train, or barge. Therefore, this change does not affect the supply of EDG fuel oil being maintained at Waterford 3. This supply of fuel oil is sufficient to power the ESF systems required to mitigate design basis accidents. A previous TS Amendment addressed the Waterford 3 FOSTs not containing enough fuel oil for testing.

The current feed tank design with the suction from the bottom instead of on the side as required by ANSI N195–1976 will not significantly decrease the margin of safety. Waterford 3 has not experienced particulate or water accumulation in the feed tanks. The fuel oil in the tank is essentially turned-over every 31 days during the EDG monthly SR, and TS Surveillances ensure water and sediment content are verified. Additionally, particulate testing is performed on the EDG FOSTs using a test filter with a smaller micron size than is on the engine. This will assure the EDG engine is not subject to failures due to particulate or water accumulation in the feed tanks.

The request for deviation from the ANSI N195–1976 requirement for the feed tank overflow to discharge to the FOST will not significantly decrease the margin of safety. The feed tank is equipped with two safety measures that would have to fail in order to allow a loss of EDG fuel oil due to over-filling a feed tank. A failure of these safety measures (high level switch to stop the transfer pump and a high level alarm in the feed tank) occurring simultaneously is very remote.

The request for deviation from ANSI N195–1976 to have one pressure indicator located at the discharge of the fuel oil transfer pump will not significantly decrease the margin of safety. A pressure indicator on the discharge of the transfer pump could indicate performance degradation of the pump. If a failure of the transfer pump occurred, indication would appear in the Control Room via the alarm for low feed tank level. The alarm for low feed tank is adequate to alert the control room of a transfer pump malfunction. However, if the transfer pump were to malfunction, the other transfer pump would be available to deliver fuel oil to operate one EDG for at least seven days. ASME Section XI testing is performed on the transfer pump once per quarter (temporary pressure instrumentation is installed on the discharge of the pump to measure pump differential pressure) to verify that pump performance has not degraded. In addition, the transfer pumps are functionally tested every month during routine testing of the EDGs.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 27, 1999.

Description of amendment request: The proposed change to the Technical Specifications (TSs), if approved, will clarify several administrative requirements, delete redundant requirements, and correct typographical errors. These revisions affect TS Sections 3.8.3.1, 3.8.3.2, 6.2.2, 6.5.1.2, 6.8.2, 6.9.1.5, and 6.9.1.6.

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 TS changes do not involve a significant increase in the probability or

consequences of an accident previously evaluated.

The changes are administrative in nature and do not impact the operation, physical configuration, or function of plant equipment or systems. The changes do not impact the initiators or assumptions of analyzed events, nor do they impact mitigation of accidents or transient events. Therefore, these changes do not increase the probability of occurrence or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature and do not alter plant configuration, require that new equipment be installed, alter assumptions made about accidents previously evaluated, or impact the operation or function of plant equipment. Therefore, these changes do not create the possibility of a new or different kind of accident than previously evaluated.

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

The proposed changes are administrative in nature and do not involve any physical changes to plant structures, systems, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions of operation, or design parameters for any SSC. The proposed changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis. Therefore, these changes do not involve any 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: J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 1, 1999.

Description of amendment request: The proposed amendments would revise the minimum fuel oil level for the diesel generator day tanks in Surveillance Requirement 3.8.1.3 and would change the acceptable fuel oil level storage band in Required Action Statement B of Limiting Condition for Operation 3.8.3.

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

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

The diesel generators are designed to supply power to the emergency systems needed to mitigate the consequences of design basis accidents such as LOCA/LOSP [loss-of-coolant accident/loss-of-offsite power]. They (the diesel generators) do not function to prevent accidents. Reducing the level requirement in the day tanks and raising the level requirement in the fuel oil storage tanks will therefore not increase the probability of occurrence of a LOCA/LOSP event. Furthermore, this proposed change does not affect any other system or piece of equipment designed to prevent the occurrence of any other design basis accident or transient. Therefore, reducing the required level in the day tanks and raising the level in the fuel oil storage tanks will not increase the probability of occurrence of any previously evaluated accident or transient.

The consequences of previously evaluated events will not be significantly increased because, with the 500-gallon day tank requirement and the increased storage tank supply, ample fuel will be available to supply the diesel generators for the duration of a LOCA/LOSP event or a station blackout event. Therefore, the consequences of an accident previously evaluated are not increased by this modification.

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

Lowering TS SR 3.8.1.3 from [greater than or equal to] 900 gallons to [greater than or equal to] 500 gallons and raising TS SR 3.8.3.1 from [greater than or equal to] 33,000 gallons to [greater than or equal to] 33,320 gallons will have no impact on the normal or emergency operation of the diesel generator and its support systems. For example, diesel generator transfer pumps and supply tank transfer pumps will continue to perform as necessary to insure an adequate supply in the respective tanks for accident mitigation.

As a result, since no new unanalyzed modes of operation are introduced, the possibility of a new or different type of accident, from any previously evaluated is not introduced.

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

The Bases for TS SR 3.8.1.3 states that the day tank must carry enough fuel oil to provide for one hour of operation, plus a 10 percent margin. This requirement is based on ANSI N195–1976 (Section 6.1).

The present 900-gallon requirement in the present Technical Specifications provides for 3.5 hours of continuous operation. Reducing the volume requirement to 500 gallons will continue to provide ample margin above the 1-hour requirement. In fact, 500 gallons in the day tank provides for 1.89 hours of continuous operation.

The Bases for TS SR 3.8.3.1 states that the fuel in the storage tanks (33,000 gallons) alone is sufficient to account for seven days of continuous operation. This is true for 33,000 gallons of *usable* fuel. However, each storage tank contains approximately 1,438 gallons of unusable fuel. Additionally, part of the current design bases for the emergency diesel generators is the ability to run four of the five diesels continuously for seven days at a load of 3250 kW. With 500 gallons in each of the four diesel's day tanks and 33,320 gallons in each of the five storage tanks, the system is capable of running continuously for 7 days. Ample onsite fuel capacity remains to operate the diesels continuously for a longer period than required to replenish the supply from outside sources. For the above reasons, the margin of safety is not significantly reduced.

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

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

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of amendment request: October 15, 1999.

Description of amendment request: The proposed amendment would change the Safety Limit Minimum Critical Power Ratios (SLMCPR) in Technical Specification (TS) 2.1.1.2 to reflect results of a cycle-specific calculation performed for Unit 1 Operating Cycle 19. The calculation was done using the new NRC-approved methodology for determining SLMCPRs.

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 Technical Specification changes do not involve a significant increase in the probability of an accident previously evaluated.

The derivation of the revised SLMCPRs for Plant Hatch Unit 1 Cycle 19 for incorporation into the TS, and their use to determine cycle-specific thermal limits, have been performed using NRC-approved methods and procedures. The procedures incorporate cycle-specific parameters and reduced power distribution uncertainties in the determination of the lower value for SLMCPRs. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from a revised method of analysis for the Unit 1 Cycle 19 core reload. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

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

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

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

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia.

Date of amendment request: October 15, 1999.

Description of amendment request: The proposed amendment would change the Safety Limit Minimum Critical Power Ratios (SLMCPR) in Technical Specification (TS) 2.1.1.2 to reflect results of a cycle-specific calculation performed for Unit 2 Operating Cycle 16. The calculation was performed using the new NRC-approved methodology for determining SLMCPRs.

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 Technical Specification changes do not involve a significant increase in the probability of an accident previously evaluated.

The derivation of the revised SLMCPRs for Plant Hatch Unit 2 Cycle 16 for incorporation into the TS, and their use to determine cycle-specific thermal limits, have been performed using NRC-approved methods and procedures. The procedures incorporate cycle-specific parameters and reduced power distribution uncertainties in the determination of the lower value for SLMCPRs. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from a revised method of analysis for the Unit 2 Cycle 16 core reload. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

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

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: October 18, 1999.

Description of amendment request: The proposed amendment would revise the activated charcoal testing methodology in accordance with the guidance provided in NRC 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. Will the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The Standby Gas Treatment (SBGT) system is used to support mitigation of the consequences of postulated accidents. The SBGT system is not considered an initiator of any analyzed accident. There is no change in function or operation of the system. The proposed change only revises the charcoal laboratory testing protocol to a more current standard that is more reliable, accurate and conservative. The change in relative humidity proposed is likewise in accordance with accepted guidance and reflective of the Vermont Yankee system configuration, which utilizes heaters to reduce the incoming humidity. The change in iodide removal efficiency is also more conservative.

Thus, the probability or consequences of previously analyzed accidents is not significantly increased.

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

This change does not affect the design or mode of operation of any plant system, structure or component. No physical alteration of plant structures, systems or components is involved and no new or different equipment will be installed. The proposed change only modifies the laboratory testing protocol and acceptance criteria to a more currently accepted standard.

Thus, the proposed change does not create the possibility of a new or different [kind of] accident from those previously evaluated.

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

The proposed changes in laboratory test protocol do not adversely affect the operation of any systems, structures or components. In fact, adopting the newer test standard will provide greater assurance that the charcoal will perform its intended function of accident consequence mitigation.

Thus, the proposed change does not significantly reduce a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 21, 1999.

Description of amendment request: The proposed amendment makes editorial and administrative changes to the Technical Specifications (TSs) by correcting two administrative errors and changing the designation of a TS-referenced figure. These changes do not materially change the meaning or application of any TS requirement.

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

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

The proposed changes are administrative or editorial in nature and do not involve any physical changes to the plant. The administrative changes do not materially affect any existing technical requirement and do not reduce the actions that are currently taken to ensure operability of plant structures, systems or components.

The changes correct past administrative errors and change a reference in the Technical Specifications and do not revise the methods of plant operation which could increase the probability or consequences of previously evaluated accidents. No new modes of operation are introduced by the proposed changes such that a previously evaluated accident is more likely to occur or more adverse consequences would result.

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

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

These changes are administrative in nature and do not affect the operation of any systems or components, nor do they involve any potential initiating events that would create any new or different kind of accident. There are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and maintained.

The changes do not affect assumptions contained in plant safety analyses or the physical design and/or modes of plant operation. Consequently, no new failure mode is introduced due to the administrative changes.

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

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

There are no changes being made to the Technical Specification safety limits or safety system settings. The operating limits and functional capabilities of systems, structures, and components are unchanged as a result of these administrative changes. These proposed changes do not affect any equipment involved in potential initiating events or safety limits. There is no change to the basis for any Technical Specification that is related to the establishment of, or the maintenance of, a nuclear safety margin.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 5, 1999.

Description of amendment request: The proposed amendments would make changes to the Technical Specifications (TSs) that are necessary to eliminate inconsistencies in the TSs pertaining to decay heat removal requirements (TSs 15.3.1.A.3, 15.3.3.A, and 15.3.3.C). An additional change to the requirements in TS 15.3.1.A.4 for pressurizer safety valve operability is also proposed to provide appropriate coordination with low temperature overpressure protection requirements. Bases revisions are provided consistent with the proposed amendments and to administratively correct references related to accumulator operability in the Bases for TS 15.3.3.

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

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

Technical Specifications 15.3.1.A.3, 15.3.3.A.3 and 15.3.3.C are all interrelated in that they each provide direction for required decay heat removal capability, either directly or indirectly by providing requirements for both support and supported systems. TS 15.3.1.A.3 provides requirements for the operation of the reactor coolant system loops, steam generators, reactor coolant pumps and residual heat removal loops as necessary to support decay heat removal from a shutdown unit. TS 15.3.3.A provides requirements for operation of the high head safety injection and low head residual heat removal system. Specifically, TS 15.3.3.A.3 provides requirements for inoperability of the residual heat removal system which accounts for the dual purpose of injection and decay heat removal. TS 15.3.3.C.2 provides requirements for operation of the Component Cooling Water System, a primary support system for both Residual Heat Removal System and Reactor Coolant Pump operation. The proposed Specifications require redundancy of decay heat removal and require placing the plant in a safe condition, maximizing the availability of decay heat removal methods when redundancy is lost. Appropriate allowances and actions are required to ensure uniform mixing of boron for reactivity control with the unit shutdown and provide for appropriate allowances to facilitate surveillance testing, and refueling operations. The time limits placed on all actions are consistent with safe operations, industry and NRC guidance. Therefore the probability of a loss of shutdown cooling or loss of subcooling; or a loss of shutdown reactivity control is minimized.

Amendments are also proposed to provide for coordination of Pressurizer Safety Valve and Pressurizer Power Operated Relief Valve operability requirements to ensure redundant overpressure protection is provided for all operating conditions. Proposed actions for inoperability of Pressurizer Safety Valves minimizes the time in that condition. Operation of the valves is not changed. Thus, the probability of a loss of coolant due to inadvertent opening of the valves is not increased. In addition, overpressure protection is maintained under all conditions such that the probability of an overpressure due to an analyzed event is not increased.

The proposed changes do not affect potential leakage paths for radiation to the environment, or of key safety barriers, and ensure appropriate system and function redundancy is maintained. Therefore the consequences of an accident previously evaluated will not increase.

Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendments do not alter the operation or method of function of the Residual Heat Removal System, Component Cooling Water System, Pressurizer Safety Valves, or Power Operated Relief Valves. The amendments provide for consistency of decay heat removal and pressure relief requirements within the Specifications providing assurance these functions can be maintained during all required plant conditions. Operations are not altered in any way that could introduce a new accident initiator not previously considered in the PBNP Safety Analyses. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments cannot create the possibility of a new or different kind of accident than any previously evaluated.

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

The proposed amendments ensure redundancy of the decay heat removal and overpressure protection over the complete range of operating conditions. Limitations are provided to ensure timely action to restore the functions to an operable condition consistent with their importance to safety. Appropriate allowances and actions are required to ensure uniform mixing of boron for reactivity control with the unit shutdown and provide for appropriate allowances to facilitate surveillance testing, and refueling operations consistent with overall safety. The functions or method of function of the systems or components affected are not being altered. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments cannot result in a significant reduction in a margin of safety.

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

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: October 21, 1999.

Description of amendment request: The request proposes to revise Technical Specification (TS) 3.4.10, Pressurizer Safety Valves (PSV), of the improved Technical Specifications issued March 31, 1999. The proposed revision is to reduce the safety valve set pressure in Limiting Condition for Operation (LCO) 3.4.10, and increase the setpoint tolerance in Surveillance Requirement (SR) 3.4.10.1. The PSV setpoint and setpoint tolerance is proposed to be changed from 2485 psig plus or minus 1% to 2460 psig plus or minus 2% in the LCO. The tolerance of plus or minus 1% in the SR is for resetting the setpoint after testing, if this is needed. The licensee also submitted the Bases pages for TS 3.4.10, which show modifications to reflect the changes to the TSs.

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

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

Any evaluations performed on an overpressure transient conservatively assume the upper limit of the pressurizer safety valve (PSV) tolerance as the pressure to which the reactor coolant system (RCS) is subjected. The proposed change to the lower tolerance limit of the pressure set point means that an overpressure transient may be terminated at a pressure that is lower than assumed in the analysis. It has also been determined that the design transients are not adversely affected because the limiting transients are not sensitive to the pressure tolerance decrease. Therefore, the primary system pressure boundary is not challenged by the PSV lower tolerance limit change. The change in the upper limit of the PSV tolerance does not challenge the upper limit of the overpressure protection. The maximum opening set pressure is not changed, and therefore, does not impact analyses performed for overpressure transients. Although the lower PSV set point would result in a lower qualified valve flow rate, the slightly lower valve flow rate would be more than compensated for by the reduced valve opening pressure. The change to the PSV set point and set point tolerance does not change the conclusions of the existing thermal hydraulic analysis for the pressurizer safety and relief system. The design function of the valves is not being changed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the USAR [Wolf Creek Updated Safety Analysis Report].

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 would allow the PSV minimum actuation pressure to be as low as 2411 psig. The pressurizer power-operated relief valve (PORV) actuation set point is 2335 psig. Therefore, the margin between the PORV and PSV actuation set points could be as low as 76 psi, which is a reduction of 49 psi from the current 125 psi margin. Even with the 30 psi pressure control uncertainty, the actuation set point margin of 76 psi is considered adequate and the PORVs are expected to continue to actuate before the PSVs during Condition 1 transients. As such, the proposed change will not have any adverse effect on the control systems. Except for the reduced lower set point, the design and operation of the PSVs are not being changed. The maximum opening pressure is not being changed. The only effect of this change would be that the PSVs could open at a lower pressure, but still above the PORV actuation set point. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The PSVs provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The PSVs are designed to prevent the system pressure from exceeding the system safety limit, 2735 psig, which is 110% of the design pressure. The change in the upper limit of the PSV tolerance from plus or minus 1% to plus or minus 2% with a reduction in the nominal set point from 2485 psig to 2460 psig does not challenge the upper limit of the overpressure protection. The maximum opening pressure set point is not changed, and therefore, does not impact analyses performed for overpressure transients. The change to PSV set point and set point tolerance does not change the conclusions of the existing thermal hydraulic analysis for the pressurizer safety and relief system. For all non-LOCA [non-loss of coolant accident] events the analyses support the change in PSV set point and set point tolerance from 2485 psig plus or minus 1% to 2460 psig plus or minus 2%. The change in the PSV set point and set point tolerance also has no effect on the Reactor Protection or Engineered Safety Features Systems trip set points. Thus, the proposed change does not involve a significant reduction in any margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

Indiana Michigan Power Company, Docket No. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: September 23, 1999, as supplemented October 11, 1999.

Brief description of amendment request: The proposed amendments involve movement of loads in excess of the design-basis seismic capability of the auxiliary building load handling equipment and structures. The proposed amendment requests approval to move the steam generator sections through the auxiliary building and to disengage crane travel interlocks, and also requests relief from performance of Technical Specification Surveillance Requirement 4.9.7.1.

Date of publication of individual notice in **Federal Register**: October 26, 1999 (64 FR 57665).

Expiration date of individual notice: November 26, 1999.

Indiana Michigan Power Company, Docket No. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: October 1, 1999.

Brief description of amendment request: The proposed amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. In addition, the licensee proposes to change the Technical Specifications dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/ hydrogen skimmer fans.

Date of publication of individual notice in **Federal Register**: October 29, 1999 (64 FR 58458).

Expiration date of individual notice: November 29, 1999.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of application for amendment: April 21, 1999, as supplemented October 15, 1999.

Brief description of amendment: The amendment allows for a one-time extension of the reactor protection system and engineered safety features actuation system instruments.

Date of issuance: October 29, 1999.

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

Amendment No.: 205.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes October 14, 1999 (64 FR 55777). The October 15, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 28, 1999, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated October 29, 1999.

Attorney for licensee: Mr. Brent L. Brandenburg, Assistant General Counsel, Consolidated Edison Company of New York, Inc., 4 Irving Place—1822, New York, NY 10003.

NRC Section Chief: Sheri Peterson.

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

Date of application for amendments: August 4, 1999.

Brief description of amendments: The amendments revise the TS (Appendix A of the Catawba operating licenses) to: (1) modify Section 3.3.2 regarding the Nuclear Service Water System, and (2) Section 5.3.1 regarding operating personnel qualifications.

Date of issuance: November 2, 1999.

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

Amendment Nos.: Unit 1–181; Unit 2–173.

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

Date of initial notice in **Federal Register**: September 8, 1999 (64 FR 48861).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 1999.

No significant hazards consideration comments received: No

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

Date of application for amendment: October 22, 1997.

Brief description of amendment: This amendment approves a proposed modification that changes the Perry facility as described in the Updated Safety Analysis Report. The change incorporates temperature control valves and associated bypass lines around the Emergency Closed Cooling system heat exchangers. These features are designed to ensure operability of the Control Complex Chilled Water System under post-accident load conditions, without the need for compensatory measures.

Date of issuance: October 29, 1999.

Effective date: October 29, 1999.

Amendment No.: 107.

Facility Operating License No. NPF–58: This amendment authorizes the revision of the Updated Safety Analysis Report.

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59922).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 1999.

No significant hazards consideration comments received: No

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

Date of application for amendments: May 18, 1999, as supplemented by letter dated September 22, 1999.

Brief description of amendments: The amendments revised Surveillance Requirements (SR) 3.8.1.3 and 3.8.1.13 to reduce the loading requirements for the emergency diesel generators (EDGs). Revised SR 3.8.1.3 requires the EDGs be loaded and operated for [greater than or equal to] 60 minutes at a load [greater than or equal to] 6500 kW and [less than or equal to] 7000 kW at least every 31 days. Revised SR 3.8.1.13 requires the EDGs to be loaded [greater than or equal to] 6900kW and [less than or equal to] 7700 kW and operated as close as practicable to 3390 kVA for 2 hours. For the remaining hours of the test, the EDGs would be loaded [greater than or equal to] 6500 kW and [less than or equal to] 7000 kW and operated as close as practicable to 3390 kVA.

Date of issuance: October 25, 1999.

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

Amendment Nos.: Unit 1–109; Unit 2–87.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 11, 1999 (64 FR 43780) The supplemental letter dated September 22, 1999, provided clarifying information that did not change the scope of the May 18, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 1999.

No significant hazards consideration comments received: No

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

Date of application for amendments: April 28, 1999.

Brief Description of amendments: These amendments revise TS Section 3.4.A.4 for Units 1 and 2. The changes relax the minimum volume requirement for the refueling water Chemical Addition Tank (CAT) from 4200 gallons to 3930 gallons. A minor administrative change is also being made to TS Table 4.1–2B to correct an earlier printing error and to delete a reference which no longer applies.

Date of issuance: November 1, 1999.

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

Amendment Nos.: 222 and 222.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register**: September 8, 1999 (64 FR 48869).

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

No significant hazards consideration comments received: No

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By December 17, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

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

Date of application for amendment: March 26, 1999, as supplemented October 15, 1999.

Brief description of amendment: The amendment allows for a one-time extension of system functional tests. The test intervals are extended for 37 months to coincide with the next refueling outage scheduled to commence on June 3, 2000.

Date of issuance: October 29, 1999.

Effective date: As of the date of issuance to be implemented upon receipt.

Amendment No.: 204.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Press release issued requesting comments as to proposed no significant hazards consideration: Yes, October 22 and 24, 1999, *Peekskill Evening Star.*

The October 15, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 28, 1999, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated October 29, 1999.

Attorney for licensee: Mr. Brent L. Brandenburg, Assistant General Counsel, Consolidated Edison Company of New York, Inc., 4 Irving Place—1822, New York, NY 10003 *NRC Section Chief*: Sheri Peterson.

Dated at Rockville, Maryland, this 9th day of November 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–29846 Filed 11–16–99; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Cumulative Report on Rescissions and Deferrals

September 1, 1999.

This report is submitted in fulfillment of the requirement of Section 1014(e) of the Congressional Budget and Impoundment Control Act of 1974 (Public Law 93–344). Section 1014(e) requires a monthly report listing all budget authority for the current fiscal year for which, as of the first day of the month, a special message had been transmitted to Congress.

This report gives the status, as of September 1, 1999, of three rescission proposals and three deferrals contained in three special messages for FY 1999. These messages were transmitted to Congress on October 22, 1998, February 1, 1999, and August 2, 1999.

Rescissions (Attachments A and C)

As of September 1, 1999, three rescission proposals totaling $35 million have been transmitted to the Congress. Attachment C shows the status of the FY 1999 rescission proposals.

Deferrals (Attachments B and D)

As of September 1, 1999, $347 million in budget authority was being deferred from obligation. Attachment D shows the status of each deferral reported during FY 1999.

Information From Special Messages

The special messages containing information on the rescission proposals and deferrals that are covered by this cumulative report are printed in the editions of the **Federal Register** cited below:

63 FR 63949, Tuesday, November 17, 1998

64 FR 6721, Wednesday, February 10, 1999

64 FR 43785, Wednesday, August 11, 1999

Jacob J. Lew,

Director.

ATTACHMENT A—STATUS OF FY 1999 RESCISSIONS

[In Millions of Dollars]

	Budgetary resources
Rescissions proposed by the President	35.0
Rejected by the Congress	
Amounts rescinded by Pub. L. 106–31, the FY 1999 Emergency Supplemental Appropriations and Rescissions Act	−16.8
Pending before the Congress for more than 45 days (available for obligation)	−18.2
Currently before the Congress for less than 45 days	

ATTACHMENT B—STATUS OF FY 1999 DEFERRALS

[In Millions of Dollars]

	Budgetary resources
Deferrals proposed by the President	1,753.0
Routine Executive releases through August 1999	−1,405.7
(OMB/Agency releases of $1,647.3 million, partially offset by a cumulative positive adjustment of $241.6 million)	