regarding when the spent fuel storage pool at MP3 will no longer be capable of supporting a full core off-load. MP3 will continue to have full core off-load capability until after refueling outage 7, currently scheduled for early calendar year 2001.

The first paragraph under ''The Need for the Proposed Action'' is changed to read:

The Need for the Proposed Action

An increase in spent fuel storage capacity is needed to maintain the capability for a full core off-load. [[Loss of full core off-load capability will occur as a result of refueling outage 7 (RFO 7), that is scheduled to start early in calendar year 2001.]] The licensee plans to install an additional 15 high density storage racks (with the capacity to store 1,104 fuel assemblies) following RFO 6 (14 will be installed between RFO 6 and RFO 7, with the last one to be installed later if it is necessary), while keeping the existing racks in place. The additional capacity will increase the capability for a full core off-load as the unit approaches the end of its operating license (November 25, 2025).

Similarly, the first paragraph under ''Reduction of Spent Fuel Generation'' is changed to read:

Reduction of Spent Fuel Generation

Generally, improved usage of the fuel and/or operation at a reduced power level would be an alternative that would decrease the amount of fuel being stored in the pool and thus increase the amount of time before full core off-load capacity is lost. With extended burnup of fuel assemblies, the fuel cycle would be extended and fewer off-loads would be necessary. [[This is not an alternative for resolving the loss of full core off-load capability because the spent fuel pool currently has the capacity for only one more full core off-load and some of the fuel to be off-loaded following RFO 7, currently scheduled for early in calendar year 2001, will have completed its operating history in the core. With the additional fuel left in the spent fuel pool after RFO 7, MP3 will no longer have the capability to conduct a full core off-load.]] Operating the plant at a reduced power level would not make effective use of available resources, and would cause unnecessary economic hardship on the licensee and its customers. Therefore, reducing the amount of spent fuel generated by increasing burnup further or reducing power is not considered a practical alternative.

Agencies and Persons Contacted

In accordance with its stated policy, on October 8, 1999, the staff consulted with the Connecticut State official, Mr. Denny Galloway of the Department of Environmental Protection, regarding the correction of the environmental assessment for the proposed action. The State official had no comments.

For further details with respect to the proposed action, see the licensee's letter dated March 19, 1999, which is available for public inspection at the U.S. Nuclear Regulatory Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC. Publically available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http:www.nrc.gov* (the Electronic Reading Room).

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

For the Nuclear Regulatory Commission.

James W. Clifford,

Chief, Section 2, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–32489 Filed 12–14–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 20, 1999, through December 3, 1999. The last biweekly notice was published on December 1, 1999 (64 FR 67330).

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

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

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

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

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

By January 14, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. 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)

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: November 18, 1999.

Description of amendment request: The proposed amendment revises the Unit 1 Heatup Curve (Technical Specification Figure 3.4.3–1), Unit 1 Cooldown Curve (Technical Specification Figure 3.4.3–2), and Unit 1 Maximum Power-Operated Relief Valve (PORV) Opening Pressure vs Temperature Curve (Technical Specification Figure 3.4.12–1) to change fluence level from 2.61×10^{19} n/cm ^{2} to 4.49×1019 n/cm ^{2} (E>1MeV). This change reflects the new actual fluence level for which these curves are valid, and is necessary to extend the applicability of the curves for Unit 1 operation.

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

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

In accordance with 10 CFR Part 50, Appendix G, the Calvert Cliffs pressure/temperature (P–T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials were developed using the methods of linear elastic fracture mechanics and the guidance found in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G. The Calvert Cliffs (P–T) limits are based on fluence level. The fluence level corresponds to the pressurized thermal shock (PTS) screening criteria defined in 10 CFR 50.61 for the critical elements. Methods described in the Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, are used to predict the embrittlement effect of neutron irradiation on reactor vessel materials. Regulatory Guide 1.99 defines embrittlement effect in terms of adjusted reference temperatures (ART), which depends on the material property of the PTS critical element.

The proposed higher fluence level for the Technical Specification P-T limits was made possible by the identification of a new 10 CFR 50.61 critical element for fracture toughness requirements for protection against PTS events. The material properties of the new critical element resulted in an increase in fluence level from 2.61×10^{19} n/cm^2 to 4.49×1019 n/cm^2 for the ART valves calculated using the material properties of the old PTS critical element. the P–T limits analysis remain well within the conservative acceptance limits of the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Hence, with the new higher fluence level, the 10 CFR Part 50, Appendix G, requirement for adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests, for the reactor coolant pressure boundary materials, is maintained.

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

2. Would not create the possibility of a new or different type of accident from any accidents previously evaluated.

The implementation of the proposed revision has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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

3. Would not involve a significant reduction in a margin of safety.

As discussed above, the P-T limits analysis remain well within the conservative acceptance limits of the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Hence, with the new higher fluence level, the 10 CFR Part 50, Appendix G, requirement for adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests, for the reactor coolant pressure boundary materials, is maintained.

Therefore, this proposed modification does not significantly reduce the margin of safety.

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

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

NRC Acting Section Chief: Victor Nerses.

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

Date of amendments request: November 19, 1999.

Description of amendments request: The amendments request approval of changes in the Updated Final Safety Analysis Report (UFSAR) that constitute an unreviewed safety question (USQ) as described in 10 CFR 50.59. Specifically, these changes would be an increase in the probability of occurrence of malfunction. Additionally, these changes were not previously evaluated in the UFSAR.

Regulations require that structures, systems, and components important to safety be appropriately protected against the effects of effects of missiles that might result from equipment failures. Failures that could occur in the large turbines of the main turbine-generator sets have the potential for producing large high-energy missiles (hereinafter called ''turbine missiles''). Both of Baltimore Gas and Electric Company's (BGE) turbine generator suppliers studied the failure of the rotating elements of their turbine-generators. The UFSAR only addresses a turbine missile hitting the Containment Building, Control Room, Switchgear Room, and Waste Processing Area. As a result of revising the Unit 1 and Unit 2 turbine missile analysis, BGE determined that the discussion of turbine missiles in Section 5.3.1 of the UFSAR was incomplete. Specifically, it did not discuss the probability of a missile from the Unit 1 turbine-generator striking: 1) the refueling water tanks; 2) the No. 11 Fuel Oil Storage Tank; or 3) plant equipment through various roof slabs or through non-missile-proof openings in the missile-proof walls. When these additional targets are included, the total target area is increased. If the target area increases, the probability of a turbine missile causing equipment damage increases. It is this increase in probability that leads to a USQ for a turbine missile from Unit 1. Note that by using methodologies previously approved by NRC, the revised analysis concludes there is no USQ for turbine missiles from the Unit 2 turbine-generator.

The UFSAR change is considered a USQ for Units 1 and 2 because the results of the revised Unit 1 turbine missile analysis for the following unprotected rooms or components show an increase in probability of occurrence of malfunction not previously evaluated in the UFSAR:

the Refueling Water Tanks;

the No. 11 Fuel Oil Storage Tank (non-missile-proof);

the saltwater pumps through roof hatches in the Intake Structure roof;

the roof slabs over the refueling Water Tank Pump Room, the Control Room Heating, Ventilation, and Air Conditioning (HVAC) Equipment Room, the Spent Fuel Pool Area Ventilation Equipment room, and a portion of 118• level roof over the fuel cask handling area;

the Control Room HVAC Room through its non-missile-proof door; and

the Unit 1 Auxiliary Building 45• Switchgear Room through the its non-missile-proof doors.

The probability of a missile from the Unit 1 turbine-generator striking them is a negligible increase in the probability of occurrence of malfunction of equipment associated with Unit 1 and 2. Upon approval of this request, the UFSAR will be revised to reflect the proposed turbine missile description. There is no USQ associated with the Unit 2 turbine-generator.

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

Regulations require that structures, systems, and components important to safety be appropriately protected against the effects of missiles that might result from equipment failures. Further that could occur in the large turbines of the main turbine-generator sets have the potential for producing large high-energy missiles (hereinafter called turbine missiles). Both of our turbine-generator suppliers studied the failure of the rotating elements of their turbine-generators. The UFSAR only addresses turbine missile hitting the Containment Building, Control Room, Switchgear Room, and Waste Processing Area. As result of revising the Unit 1 and Unit 2 turbine missile analysis, we determined that the discussion of turbine missiles of the UFSAR was incomplete. From the revised analysis, we determined Unit 1 and 2 USQs exist for the following unprotected rooms or components (i.e., there is an increase in probability of occurrence of malfunction not previously evaluated in the UFSAR):

the Refueling Water Tanks;

the No. 11 Fuel Oil Storage Tank;

the Saltwater Pumps through roof hatches in the Intake Structure Roof;

the roof slabs over the Refueling Water Tank Pump Room, the Control Room Heating, Ventilation, and Air Conditioning (HVAC) Equipment Room, Spent Fuel Pool Area Ventilation Equipment Room, and a portion of 118′ level roof over the cask handling area;

the Control Room HVAC Room through its non-missile-proof door; and,

the Unit 1 Auxiliary Building 45′ Switchgear Room through its non-missile-proof doors.

The probability of a missile from the Unit 1 turbine-generator striking them is a negligible, but greater than zero, increase in the probability of occurrence of malfunction of equipment associated with Units 1 and 2.

For Unit 1 High Trajectory Missiles (HTM), the guidance of NUREG 0800, Standard Review Plan, is used as one acceptable method for evaluating the risk. Use of this method is not a commitment to the Standard Review Plan and does not incorporate the Standard Review Plan into our licensing basis. The revised analysis shows that the total target area considered vulnerable to an HTM is less than the Standard Review Plan limit of 10,000 ft^2 for each unit. Therefore, the risk form an HTM is insignificant. Note that all of the Units 1, 2, and Common structures listed above are equally vulnerable to a Unit 1 HTM. Therefore, any risk increase to the plant structures constitutes a USQ for Units 1 and 2.

For Unit 1 Low-Trajectory Missiles (LTMs), protection for the Auxiliary Building is provided by a 3' thick, concrete, missile-proof wall between the Turbine Building and the Auxiliary building (the K-line wall). This wall is 3' thick below the 69' elevation and 2' thick above the 69' for areas protecting safety-related equipment. The revised analysis evaluates the protection of Unit 1 equipment from a Unit 1 LTM. The 69' Control Room HVAC Equipment Room and Unit 1 Auxiliary Building 45' Switchgear Room are protected by the missile-proof walls except for the openings at the non-missile-proof doors. A turbine missile that hits one of these doors is assumed to go through them, strike safety-related equipment in the room, and cause it to fail. Recall that the Control Room HVAC equipment is shared by both units. Therefore, any increase in risk of failure of equipment in this room affects both Units 1 and 2.

The risk associated with a turbine missile to either of these doors is calculated using guidance in Regulatory Guide 1.115, Revision 1, ''Protection Against Low-Trajectory Turbine Missiles.'' This guidance states that the turbine missile hazard should be less than 10^7. The missile hazard rate in the revised risk analysis shows that the risk from LTMs from the Unit 1 General Electric turbine-generator to the 69' Control Room HVAC Equipment Room and Unit 1 Auxiliary Building 45' Switchgear room through these non-missile-proof doors is less than 10^7.

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

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed change makes no physical changes to the plant. Specifically, the proposed change does not add new or modify existing plant equipment such that it could become an accident initiator different from its current role as an accident initiator. The only change made by this activity is the revision of the UFSAR to include the revised turbine missile analysis. The UFSAR chapter 1 drawings correctly depict the location of plant structures and components, including the thickness of and the openings in the missile-proof wall between the Turbine Building and the Auxiliary building (the K-Line Wall). Therefore, the possibility of a new or different type of accident is not created by the proposed change.

3. Would not involve a significant reduction in a margin of safety.

The regulations require an evaluation of turbine missiles to ensure that structures, systems, and components important to safety be appropriately protected from them. Revised turbine missile analysis have been performed consistent with appropriate regulatory guidance (Regulatory Guide 1.115 and the Standard Review Plan). The results of the revised analysis meet the acceptance criteria of the guidance. 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 amendments request involves no significant hazards consideration.

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

NRC Acting Section Chief: Victor Nerses.

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

Date of amendment request: November 17, 1999.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 2.1.1.2, ''Reactor Core Safety Limits.'' The minimum critical power ratios (MCPR) for single and two recirculation loop operation would be increased. In addition, the reference in TS 5.6.5, ''Core Operating Limits Report,'' Item b.5, would be removed.

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 license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed license amendment will establish MCPR Safety Limit values of 1.10 for two recirculation loop operation and 1.11 for single recirculation loop operation. Additionally, the proposed license amendment replaces an expiring cycle-specific reference in the list of analytical methods approved for determining core operating limits in Specification 5.6.5.b with a reference to a GE [General Electric] topical report which has been accepted by the NRC.

The methods for calculating the MCPR Safety Limit values have been previously approved by the NRC and are described in GE's reload licensing methodology topical report NEDE–24011–P–A. Use of these methods ensures that the integrity of the fuel will be maintained during normal operation and that the resulting MCPR Safety Limit values satisfy the fuel design safety criteria that less than 0.1 percent of the fuel rods experience boiling transition if the safety limits are not violated. The change does not require any physical plant modifications, physically affect any plant components, or allow the plant to be operated any closer to fuel design limits. Therefore, the proposed change to the MCPR Safety Limit values and to the list in Specification 5.6.5.b of analytical methods approved for determining core operating limits results no increase in the probability of a previously evaluated accident.

The consequences of a previously evaluated accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the accident, the availability and successful functioning of the equipment assumed to operate in response to the accident, and the setpoints at which these actions are initiated.

The methods used for calculating the MCPR Safety Limits have been approved by the NRC and are described in GE's reload licensing methodology topical report NEDE–24011, ''General Electric Standard Application for Reactor Fuel (GESTAR II).'' The proposed MCPR Safety Limit values of 1.10 for two recirculation loop operation and 1.11 for single recirculation loop operation will ensure that less than 0.1 percent of the fuel rods will experience boiling transition during any plant operation if the limits are not violated. The proposed change to the MCPR Safety Limit values does not affect the performance of any equipment used to mitigate the consequences of a previously evaluated accident. Also, the proposed change does not affect setpoints that initiate protective or mitigative actions. No analysis assumptions are violated and there are no adverse effects on the factors contributing to offsite and onsite dose.

Based on the determination of the proposed MCPR Safety Limit values using conservative NRC-approved methods and the operability of plant systems designed to mitigate the consequences of accidents not being changed, the proposed change to the MCPR Safety Limit values and to the list in Specification 5.6.5.b of analytical methods approved for determining core operating limits does not significantly increase the consequences of a previously evaluated accident.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This proposed license amendment does not involve any physical alteration of plant systems and plant equipment will not be operated in a different manner. As a result, no new failure modes are being introduced. Therefore, the proposed change to the MCPR Safety Limit values and to the list in Specification 5.6.5.b of analytical methods approved for determining core operating limits will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety.

The margin of safety is established through the design of the plant structures, systems, and components; through the parameters within which the plant is operated; through the establishment of setpoints for actuation of equipment relied upon to respond to an event; and through margins contained within the safety analyses.

The proposed change to the MCPR Safety Limit values and the list in Specification 5.6.5.b of analytical methods approved for determining core operating limits does not adversely impact the performance of plant structures, systems, components, and setpoints relied upon to respond to mitigate an accident. As previously stated, the methods for calculating the MCPR Safety Limit values have been previously approved by the NRC and are described in GE's reload licensing methodology topical report NEDE–24011–P–A. Use of these methods ensures that the resulting MCPR Safety Limit values satisfy the fuel design safety criteria that less than 0.1 percent of the fuel rods experience boiling transition if the safety limits are not violated. As a result, the proposed changes do not significantly impact any safety analysis assumptions or results. Based on the assurance that the fuel design safety criteria will be met, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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: November 19, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for the Harris Nuclear Plant (HNP) to incorporate American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' as the standard for testing nuclear-grade activated charcoal. Specifically, TS 4.7.6 will be revised for the Control Room Emergency Filtration System, TS 4.7.7 will be revised for the Reactor Auxiliary Building Emergency Exhaust System, and TS 4.9.12 will be revised for the Fuel Handling Building Emergency Exhaust System. These changes are being proposed in accordance with NRC Generic Letter (GL) 99–02, ''Laboratory Testing Of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

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

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

This proposed change to revise the standard to which activated charcoal samples are tested will ensure that testing is accurate and repeatable. This will help ensure that the Engineered Safety Feature (ESF) ventilation systems are capable of performing their safety function. Therefore, the proposed changes do 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.

The proposed changes incorporate ASTM D3803–1989 as the testing standard for nuclear-grade activated charcoal samples. This will ensure that testing is accurate and repeatable. Plant structures, systems, and components will not be operated in a different manner as a result of these proposed changes and no physical modifications to equipment are involved. Using the improved testing protocol does not have the potential for creating the possibility of a new or different type of accident from any previously evaluated.

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

The proposed changes do not change the manner in which structures, systems or components are operated. Revising the standard to which activated charcoal samples are tested will ensure that testing is accurate and repeatable. This will help ensure that the ESF ventilation systems are capable of performing their safety function. Therefore, the proposed 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 12, 1999.

Description of amendment request: The proposed change revises the pressure-temperature limits by revising the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel to a maximum of 32 Effective Full Power Years.

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes do not modify the reactor coolant pressure boundary, do not make changes in operating pressure, materials or seismic loading. The proposed changes adjust the reference temperature for the limiting beltline material to account for radiation effects and provide the same level of protection as previously evaluated. The proposed changes do not adversely affect the integrity of the reactor coolant system (RCS) such that its function in the control of radiological consequences is affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Quad Cities Nuclear Power Station. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Use of the revised pressure-temperature (P–T) curves will continue to provide the same level of protection as was previously reviewed and approved.

Further, the proposed changes to the P–T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate any accident sequence for Quad Cities Nuclear Power Station. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes reflect an update of the P–T curves to extend the Reactor Pressure Vessel (RPV) operating limit to 32 Effective Full Power Years (EFPY). The revised curves are based on the latest American Society of Mechanical Engineers (ASME) guidance and actual operational data for the units. This proposed changes are acceptable because the ASME guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME Section IX Appendix G was approved in 1974. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 16, 1999.

Description of amendment request: The proposed change modifies the surveillance requirements for Functional Unit 3 on Table 4.1.A–1 due to replacement of the Reactor Pressure Vessel Steam Dome pressure switches with analog trip units.

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

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

During the upcoming refueling outages at Quad Cities Nuclear Power Station, Unit 1 and Unit 2, a design change will be implemented that upgrades the existing Reactor Vessel Steam Dome-High instrumentation from a pressure switch to an analog trip unit device. Analog trip units are proven technology that are more reliable than existing equipment. Analog trip units are used in various applications of Quad Cities Nuclear Power Station, including the Reactor Protection System (RPS) low water level trip function.

The proposed change adds a CHANNEL CHECK and 31-day trip unit calibration requirement for the Reactor Vessel Steam Dome Pressure—High RPS trip function. This requirement is not applicable to the existing instrumentation because the Barksdale pressure switches are non-indicating and do not employ trip units.

Technical Specification (TS) requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed change does not adversely impact structures, systems, or components (SSCs). The planned instrument upgrade is a more reliable design than existing equipment. The proposed change establishes requirements that ensures components are operable when necessary for the prevention or mitigation of accidents or transients. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes support a planned instrumentation upgrade by incorporating Surveillance Requirements required to ensure operability. The change does not adversely impact the manner in which the instrument will operate under normal and abnormal operating conditions. Therefore, these changes provide an equivalent level of safety and will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change supports a planned instrumentation upgrade. The proposed change does not affect the probability of failure or availability of the affected instrumentation. The addition of a CHANNEL CHECK and 31-day trip unit calibration for RPS Functional Unit 3 (Reactor Vessel Steam Dome Pressure—High) is a conservative change that aligns the surveillance requirements for a planned instrumentation upgrade with that of similar instrumentation. Therefore, it is concluded that the proposed changes will not result in 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 13, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.6.1, Table 3.3.6–1, ''Primary Containment

Isolation Instrumentation.'' This amendment requests that Function 5 on Table 3.3.6–1, ''RHR SDC System Isolation,'' be modified by removing footnote (d). Footnote (d) states, ''Only the inboard trip system is required in Modes 1, 2, and 3, as applicable, when the outboard valve control is transferred to the alternate remote shutdown panel and the outboard valve is closed.'' The outboard suction valve, RHR–V–8, is no longer used as a high/low pressure interface in the residual heat removal (RHR) system. Valve RHR–V–9, which is in series with valve RHR–V–8, is now used as the high/low pressure interface valve. Valve RHR–V–9 is operable in all modes of operation and therefore, footnote (d) is no longer needed. The current footnote (e) will be relettered as footnote (d) for consistency.

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.

This change involves the probability and consequences of accidents associated with the isolation of the RHR SDC [shutdown cooling] mode of RHR operation. Isolation is provided if high temperatures occur in RHR pump rooms or heat exchanger areas, if reactor vessel water level is low, or if reactor vessel pressure is high.

FSAR [Final Safety Analysis Report] Chapter 15, ''Accident Analysis,'' describes two events associated with the RHR system during SDC operation. FSAR Section 15.1.6, ''Inadvertent Residual Heat Removal Shutdown Cooling Operation,'' describes the impact of system operation during startup or cool-down when the reactor is near critical. The proposed change removes the exemption for the second trip system to isolate RHR SDC operation. There will be no change in the probability or consequences of this accident as a result of the proposed change.

The second accident is described in FSAR Section 15.2.9, ''Failure of Residual Heat Removal Shutdown Cooling.'' It postulates the failure of the RHR system to function in SDC mode. The evaluation assumes a failure of the SDC mode of operation but does not disable the remaining modes of RHR operation. The alternate SDC paths involve the use of the safety relief valves to establish a cooling flow path to the containment suppression pool. That evaluated accident does not result in any fuel failure. The proposed change will not result in an increase in the probability of fuel failures. The evaluated accident does result in normal coolant activity being released to the suppression pool through the safety relief valves. The proposed activity will not result in a change in the release of this coolant activity. The proposed change requires the removal of the exemption for the second trip system to isolate SDC and will have no impact on the probability or consequences of that accident.

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

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

The proposed change will not cause any new inadvertent SDC startup, loss of water inventory or loss of coolant accidents (LOCA). New or different inadvertent RHR SDC startup accidents are not possible because this change is only a further restriction on system operation. The LOCA during Mode 3 is bounded by the LOCA defined for Modes 1 and 2. No new primary system LOCA can be initiated because of this change.

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

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

The removal of an exemption for the second trip system, as proposed by this change, will increase the probability that leaks and high pressure will be isolated. Therefore, operation of WNP–2 in accordance with the proposed amendment will not decrease the margin of safety. Therefore, the operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 25, 1999.

Description of amendment request: The proposed license amendment would revise the reactor pressure vessel (RPV) surveillance capsule withdrawal schedule for the River Bend Station. The first surveillance capsule would be withdrawn at 13.4 effective full power years (EFPY) rather than 10.4 EFPY.

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

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

Pressure-temperature (P/T) limits (RBS Technical Specifications Figure 3.4.11–1) are imposed on the reactor coolant system to ensure that adequate safety margins against nonductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P/T limits are related to the nil-ductility reference temperature, RT_{NDT}, as described in ASME [American Society of Mechanical Engineers] Section III, Appendix G. Changes in the fracture toughness properties of RPV beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR [Part] 50, Appendix H. The effect of neutron fluence on the shift in the nil-ductility reference temperature of pressure vessel steel is predicted by methods given in RG [Regulatory Guide] 1.99, [Revision] 2.

River Bend's current P/T limits, as well as those for the planned increase in reactor thermal power (''Power Uprate''), were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in RG 1.99, [Revision] 2, Regulatory Position 1. Calculation of adjusted reference temperature by these procedures includes a margin term to ensure conservative, upper-bound values are used for the calculation of the P/T limits. Revision of the first capsule withdrawal schedule will not affect the P/T limits because they will continue to be established in accordance with Regulatory Position 1 or other NRC [Nuclear Regulatory Commission]-approved procedures. When permitted (two or more credible surveillance data sets available), Regulatory Position 2 (or other NRC-approved) methods for determining adjusted reference temperature will be followed.

This change is not related to any accidents previously evaluated. The proposed change is a revision of the first surveillance capsule withdrawal time, identified in TRM [Technical Requirements Manual] Table 3.4.11–1, from 10.4 EFPY to 13.4 EFPY. This change will not affect P/T limits as given in RBS Technical Specifications Figure 3.4.11–1 or USAR Figures 5.3–4a and 5.3–4b. This change will not affect any plant safety limits or limiting conditions of operation. The proposed change will not affect reactor pressure vessel performance as no physical changes are involved and RBS vessel P/T limits will remain conservative in accordance with RG 1.99, [Revision] 2 requirements. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents. Therefore, the probability or consequences of accidents previously evaluated will not be increased by the proposed change.

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

The proposed change revises the first RPV material surveillance capsule withdrawal time in TRM Table 3.4.11–1 from 10.4 EFPY to 13.4 EFPY. This proposed change does not involve a modification of the design of plant structures, systems, or components. The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of structures, systems, or components important to safety (ITS) as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of ITS equipment will not be increased, and increased or more severe testing of ITS equipment will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from that previously evaluated.

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

As stated in Section 5.3.2 of the River Bend Safety Evaluation Report (NUREG–0989), ''Appendices G and H of 10 CFR [Part] 50 describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those commended in the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code], Section III, Appendix G. * * * Until the results from the reactor vessel surveillance program become available, the staff will use Regulatory Guide (RG) 1.99, Revision 1 [now Revision 2], to predict the amount of neutron irradiation damage.* * * The use of operating limits based on these criteria—as defined by applicable regulations, codes, and standards—will provide reasonable assurance that nonductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria (GDC) 31.''

Bases for RBS Technical Specification 3.4.11 states: ''The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB [reactor coolant pressure boundary], a condition that is unanalyzed. * * * Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.''

The proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed change does not involve revision of the P/T limits but rather a revision of the withdrawal time for the first surveillance capsule. The current P/T limits (and proposed P/T limits for Power Uprate) were established based on adjusted reference temperatures for vessel beltline materials calculated in accordance with Regulatory Position 1 of RG 1.99, [Revision] 2. P/T limits will continue to be revised as necessary for changes in adjusted reference temperature due to changes in fluence according to Regulatory Position 1 until two or more credible surveillance data sets become available. When two or more credible surveillance data sets become available, P/T limits will be revised as prescribed by Regulatory Position 2 of RG 1.99, [Revision] 2, or other NRC-approved guidance. Therefore, the proposed changes do not involve a significant reduction in any margins of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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: October 29, 1999.

Description of amendment request: The proposed license amendment would change the River Bend Station (RBS) Updated Safety Analysis Report (USAR), Sections 6.2 and 15.6, to incorporate a revision to the calculation of radiological doses following a loss-of-coolant-accident (LOCA). The LOCA dose calculation was revised as a result of (1) an increase in the calculated positive pressure period (PPP) to account for a new phenomenon identified in Information Notice (IN) 88–76, (2) a more conservative Suppression Pool water volume value, (3) an additional and more conservative liquid leakage term identified in IN 91–56, and (4) changes to the engineered safety features (ESF) systems liquid leakage term.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

The analysis changes described by this proposed change to the USAR are not initiators to events, and therefore do not involve the probability of an accident. These modifications reflect a revision to the post-LOCA dose calculation. USAR Section 15.6.5.1.1 states that ''There are no realistic, identifiable events which would result in a pipe break inside of containment of the magnitude required to cause an accident LOCA * * * However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.'' The analysis itself does not identify an initiator, nor is it the initiator, of a LOCA. There was no physical change to the plant. The increase to the positive pressure period (PPP) was the result of inclusion of phenomena not previously included in the analysis documented in the SAR [safety analysis report], and does not have any impact on accident probability. The inclusion of an NRC [Nuclear Regulatory Commission] Information Notice (IN) 91–56 unfiltered liquid leakage term is voluntary and conservative in nature and does not represent an additional failure that could be construed as an initiator to the event. Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the safety analysis report (SAR).

This proposed change to the USAR does increase the consequences of an accident, but the increase is not significant. While the calculated off-site and control room doses of a LOCA did increase in Revision 1 to the post-LOCA dose calculation (reference 1) [of Attachment 1 to the License Amendment request, dated October 29, 1999], the dose consequences remain below the regulatory limits of 10 CFR [Part] 100 and 10 CFR [Part] 50, Appendix A, General Design Criteria (GDC) 19 as approved per NUREG–0989 and License Amendment 98. This change first accounts for the potential effect that differential temperature has on the PPP assumed in the off-site dose analysis. It also conservatively includes an additional liquid leakage term to account for concerns documented in NRC IN 91–56. Neither of these changes has an appreciable effect on vital area access doses. Vital area access dose calculations were not revised since they still conservatively reflect the expected doses discussed in USAR Section 12.3.2.4. There is no impact on equipment qualification associated with the proposed change since other gross conservatisms exist in those calculations (e.g., not crediting suppression pool scrubbing) compared to the post-LOCA dose calculations. Reanalysis of the off-site dose calculation demonstrates that the revised doses are increased only slightly and remain significantly less than the regulatory limits. With the IN 91–56 term excluded, the increases are within the criteria of less than 10 [percent] of the remaining margin, which is the criteria to be applied in the revised 10 CFR 50.59 rule for minimal increases in consequences. With the IN 91–56 term included, only the 30 day LPZ [low-population zone] thyroid dose exceeds the ''minimal increase'' criterion. Note the doses documented in Table 1 [of Attachment 1 to the License Amendment request, dated October 29, 1999], above, are less than the values which had been documented in the SAR prior to the implementation and NRC approval of TS [Technical Specifications] Amendment 98. Therefore, this change does not significantly increase the consequences of an accident previously evaluated in the SAR.

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

This change does not represent a physical change to the plant. It does not involve initiators to any events in the SAR, nor does the activity create the possibility for any new accidents. Rather, this change is a result of the evaluation of the most limiting LOCA which can occur at River Bend. Therefore, this change involves no new system interactions and does not create the possibility of an accident of a different type than those presently evaluated in the SAR.

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

The off-site dose consequences are calculated in accordance with regulatory guidance found in Regulatory Guide 1.3 and the SRP [Standard Review Plan], consistent with the analyses submitted to and approved by the NRC in support of Technical Specification Amendment 98. It is conservatively assumed that 100 [percent] fuel failure occurs instantaneously upon a recirculation pipe break, thus 2 of the 3 fission product barriers are immediately eliminated. These assumptions are made without any causes for the failures being identified. Containment is assumed to leak at its maximum allowable leakage rate (0.26 [percent] per day) for the duration of the event. Other leakage terms, such as engineered safety feature (ESF) leakage, are assumed to be equal to the Technical Specification limit. Since assumptions are made in accordance with Technical Specification allowable values and regulatory guidance, this change does not reduce the margin of safety as defined in the basis for any RBS Technical Specification.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 29, 1998, as supplemented by letters dated July 29, October 28, and November 11, 1999.

Description of amendment request: The amendment will revise Technical Specification 6.9.1.11.1 by replacing the existing reference to the Asea Brown Boveri-Combustion Engineering, Inc. (ABB CE), small break loss-of-coolant (SBLOCA) accident emergency core cooling system (ECCS) performance evaluation model with the revised model described in the topical report CENPD–137, Supplement 2, P–A, April 1998.

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

1. 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 SBLOCA ECCS performance evaluation is conducted to demonstrate conformance of light water nuclear power reactors to the ECCS acceptance criteria of 10 CFR 50.46. The proposed change is associated with an analysis performed using the new Supplement 2 version of the ABB CE SBLOCA Model (S2M). The primary objective of the analysis using the new model was to determine the impact of a reduction in High Pressure Safety Injection (HPSI) pump flow rate due to increased surveillance test measurement uncertainty. NRC approval of the new S2M model for use in licensing applications of CE design pressurized water reactors was obtained on December 16, 1997 (Reference 1) [of license amendment request dated July 29, 1998].

A comparison of the Waterford 3 results for the limiting SBLOCA scenario using the new S2M model against the criteria of 10 CFR 50.46(b) is summarized below:

Parameter	Result	Criterion
Peak Cladding Temperature	1929°F	2200°F
Maximum Cladding Oxidation	8.09%	17%
Core-wide Cladding Oxidation	<0.58%	1%
Coolable Geometry Maintained	Yes	Yes

These results remain within the criteria of 10 CFR 50.46. Thus, application of the new S2M model to the ECCS at Waterford will not involve a significant increase in the probability or consequences of any 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 will not create any new system connections or interactions. Thus, no new modes of failure are introduced. The revised methods used in the new SBLOCA evaluation model and their impact has been reviewed and approved by the NRC (Reference 1) [of license amendment request dated July 29, 1998]. 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 does not alter the ability of the ECCS to maintain compliance with 10 CFR 50.46 criteria. The revised methods used in the new SBLOCA evaluation model and their impact has been reviewed and approved by the NRC (Reference 1) [of license amendment request dated July 29, 1998]. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 7, 1999.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.3.2.1, ''Safety Features

Actuation System Instrumentation,'' Table 3.3–4, ''Safety Features Actuation System Instrumentation Trip Setpoints,'' to remove the ''Trip Setpoint'' values and modify the ''Allowable Values'' for Containment Pressure-High and Containment Pressure-High-High, and would change TS 3/4.3.2, ''Reactor Protection System and Safety System Instrumentation,'' to reflect the above change.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes do not change any accident initiator, initiating condition, or assumption.

The proposed changes would revise Technical Specification (TS) Table 3.3–4, Safety Features Actuation System Instrumentation Trip Setpoints, to administratively remove from TS the ''Trip Setpoint'' values for Instrument String Functional Unit ''b'', Containment Pressure—High, and Functional Unit ''c'', Containment Pressure—High-High, and also modify the TS ''Allowable Values'' entry for these same Functional Units, consistent with updated calculations using current setpoint methodology. The Trip Setpoint values removed from TS will be maintained in DBNPS-controlled documents. The proposed changes to Limiting Condition for Operation (LCO) 3.3.2.1 and Bases 3/4.3.1 and 3/4.3.2 are associated with these changes.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path or alter radiological consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety because the proposed changes establish an error analysis that has been shown to adequately preserve the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 2, 1999.

Description of amendment request: The proposed amendment would: (1) relocate the Boric Acid Addition Tank System (BAAS) and Borated Water Storage Tank requirements of Technical Specification (TS) 3/4.1.2.8, Reactivity Control Systems—Borated Water Sources—Shutdown, in their entirety to the Davis-Besse Nuclear Power Station Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM); (2) relocate the BAAS requirements of TS 3/4.1.2.9, Reactivity Control Systems—Borated Water Sources—Operating, to the USAR TRM, except for portions applicable to the BWST which are proposed to be deleted because they are redundant to the existing provisions of TS 3/4.5.4, Emergency Core Cooling Systems—Borated Water Storage Tank; (3) modify TS 3/4.1.2.1, Reactivity Control Systems—Borated Water Sources—Shutdown, by deleting references to TS 3.1.2.8; (4) incorporate corresponding changes to the TS index; and (5) incorporate corresponding changes to the TS Bases.

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

The proposed changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions remain as previously analyzed.

The proposed changes would relocate the Boric Acid Addition System (BAAS) and Borated Water Storage Tank (BWST) requirements of Technical Specification (TS) 3/4.1.2.8 in their entirety to the Davis-Besse Nuclear Power Station (DBNPS) Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). The proposed changes would also relocate the BAAS requirements of TS 3/4.1.2.9 to the USAR TRM. The portions of TS 3/4.1.2.9 applicable to the BWST are proposed to be deleted because they are completely redundant to the existing provisions of TS 3/4.5.4, Emergency Core Cooling Systems—Borated Water Storage Tank. Associated with these changes, TS 3/4.1.2.1 is proposed to be revised to delete references to TS 3.1.2.8. The appropriate changes to the TS Index are also proposed, as well as changes to TS Bases 3/4.1.2. The proposed changes are also consistent with the improved ''Standard Technical Specifications—Babcock and Wilcox Plants,'' NUREG–1430, Revision 1.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

The chemical addition system, which includes the BAAS, is not credited for mitigation of any USAR Chapter 6 or Chapter 15 accidents. The BWST is credited for mitigation of USAR Chapter 6 and Chapter 15 accidents, as part of the Emergency Core Cooling System (ECCS). However, the BWST's requirements concerning ECCS are provided in separate TS 3/4.5.4, that is not proposed for change.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety. No new or different types of failures or accident initiators are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative in nature, consisting of deletion and/or relocation of certain TS requirements into licensee-controlled documents, and have no bearing on the margin of safety which exists in the present TS or USAR.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 2, 1999.

Description of amendment request: The proposed amendment would: (1) modify Technical Specification (TS) 3/4.3.2.1, Safety Features Actuation System Instrumentation, Table 3.3–4, Safety Features Actuation System Instrumentation Trip Setpoints, to remove ''Trip Setpoint'' values for Instrument String Functional Unit ''f,'' Borated Water Storage Tank (BWST) Level; (2) modify TS 3/4.3.2.1, Table 3.3–4, Functional Unit ''f,'' Allowable Values, to make it consistent with updated calculations using current setpoint methodology; (3) modify Limiting Condition for Operation (LCO) 3.3.2.1, Safety Features Actuation System Instrumentation to reflect removal of the ''Trip Setpoint'' for this Functional Unit; (4) change the footnote associated with TS 3/4.3.2.1, Table 3.3–4, Functional Unit ''f,'' Allowable Values, to indicate that the Allowable Values apply to the Channel Functional Test and no longer applies to the Channel Calibration; (5) modify TS 3/4.1.2.9, Reactivity Control Systems—Borated Water Sources—Operating, and TS 3/4.5.4, Emergency Core Cooling Systems—Borated Water Storage Tank, to increase the minimum BWST water level; and (6) make corresponding changes to TS Bases 3/4.1.2, Boration Systems, 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, and 3/4.5.4, Borated Water Storage Tank.

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

The proposed changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes do not change any accident initiator, initiating condition, or assumption.

The proposed changes would revise Technical Specification (TS) Table 3.3.4, Safety Features Actuation System Instrumentation Trip Setpoints, to administratively remove from the TS the ''Trip Setpoint'' values for Instrument String Functional Unit ''f,'' Borated Water Storage Tank (BWST) Level, and also modify the TS ''Allowable values entry for this same Functional Unit, consistent with updated calculations using current setpoint methodology. The Trip Setpoint values removed from the TS will be maintained in Davis-Besse Nuclear Power Station (DBNPS)-controlled documents. The proposed changes to Limiting Condition for Operation (LCO) 3.3.2.1 and Bases 3/4.3.1 and 3/4.3.2 are associated with these changes.

Associated with the above changes, TS 3/4.1.2.9 and TS 3/4.5.4 are proposed to be revised to increase the minimum available BWST borated water volume requirement as specified in LCO 3.1.2.9.b.1 and LCO 3.5.4.a. The proposed changes to Bases 3/4.1.2 and Bases 3/4.5.4 are associated with these changes. These changes are consistent with the revised setpoint analyses.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety because the proposed changes establish an error analysis that has been shown to adequately preserve the margin of safety, and the trip setpoint values removed from the TS will be maintained in the DBNPS Updated Safety Analysis Report, with proposed changes subject to the regulatory requirements of 10 CFR 50.59.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 8, 1999.

Description of amendment request: The proposed amendment would relocate Technical Specification (TS) 6.5.1, Station Review Board, and TS 6.5.2, Company Nuclear Review Board, to Davis-Besse Updated Safety Analysis Report Chapter 17.2, Quality Assurance During the Operations Phase, also known as the Quality Assurance Program. The proposed changes are consistent with the recommendations in NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance.''

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

The proposed changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are affected by the proposed changes to Section 6.0, Administrative Controls, of the Technical Specifications (TS).

The proposed changes to relocate the detailed listings of TS Section 6.5.1, Station Review Board (SRB), and TS 6.5.2, Company Nuclear Review Board (CNRB), to the Davis-Besse Nuclear Power Station (DBNPS) Quality Assurance Program in Chapter 17 of the Updated Safety Analysis Report are consistent with the NRC's guidance in NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 1 and NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance,'' dated December 12, 1995. These TS being relocated will remain subject to the controls of other NRC regulations (e.g., 10 CFR 50.54(a)). The proposed changes to the TS Index reflect the relocation of TS 6.5.1 and TS 6.5.2. These are administrative changes that do not reduce the duties or responsibilities of the SRB and CNRB in ensuring the safe operation of the DBNPS.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. As described above, these changes are consistent with the improved ''Standard Technical Specifications—Babcock and Wilcox Plants'' (NUREG–1430 Revision 1) and Administrative Letter 95-06, and are administrative changes. The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes, which involve the administrative location for listing SRB and CNRB responsibilities. The proposed changes do not alter any accident scenarios.

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative and do not reduce or adversely affect the capabilities of any plant structures, systems or components to perform their nuclear safety functions.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 1, 1999

Description of amendment request: The proposed license amendment is prescribed by the requested actions of Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The proposed amendment will modify the existing Ventilation Filter Testing Program contained in Technical Specification 5.5.7.c by replacing the reference to ASTM D3803–1986, the standard for charcoal filter testing for ESF ventilation systems, with ASTM D3803–1989. The proposed amendment will also incorporate the suggested safety factor for charcoal filter efficiency regarding methyl iodide penetration.

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

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

The proposed change to reference American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' for laboratory testing of Engineered Safety Features (ESF) ventilation systems in lieu of ASTM D3803–1986 is prescribed by the requested actions of Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The use of ASTM D3803–1989 allows for increased accuracy in monitoring the degradation of ESF ventilation system activated carbon (charcoal) over time and is a reproducible method for determining the realistic capability of charcoal. The 1989 standard is endorsed by the NRC and is considered to be more stringent regarding testing criteria than the previous referenced standard (1986). GL 99–02 encourages addressees, if necessary, to amend their Technical Specifications (TS) to reference ASTM D3803–1989 for charcoal filter laboratory testing for ESF ventilation systems. In response to the referenced GL, the proposed change modifies the existing Perry Nuclear Power Plant (PNPP) Ventilation Filter Testing Program (VFTP) contained in the PNPP TS to reference ASTM D3803–1989 as the standard for charcoal filter laboratory testing for ESF ventilation systems. In addition, the proposed change incorporates the safety factor suggested within GL 99–02 for charcoal filter efficiency with respect to methyl iodide penetration. The proposed change provides assurance for compliance with the current licensing basis regarding dose limits of General Design Criteria (GDC) 19 of Appendix A to 10 CFR 50 and 10 CFR 100. The proposed change ensures originally stated design criteria are met and therefore does not affect the precursors for accidents or transients analyzed in Chapter 15 of the PNPP Updated Safety Analysis Report (USAR). With the proposed change, the radiological consequences are the same as previously stated in the USAR. Therefore, the implementation of the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to reference ASTM D3803–1989 for the laboratory testing of charcoal filters of ESF ventilation systems in lieu of ASTM D3803–1986 is prescribed by the requested actions of GL 99–02. ASTM D3803–1989 is endorsed by the NRC and is considered a more stringent testing standard than the previous referenced standard, ASTM D3803–1986. In addition, the proposed change incorporates the safety factor suggested within GL 99–02 for charcoal filter efficiency with respect to methyl iodide penetration. The proposed change provides assurance for compliance with the current licensing basis regarding dose limits of GDC 19 of Appendix A to 10 CFR 50 and 10 CFR 100. The proposed change does not change the assumptions used in any accident analysis and no new or different kind of accident is created. The proposed change ensures originally stated design criteria are met and therefore does not affect the precursors for accidents or transients analyzed in Chapter 15 of the PNPP USAR. Therefore, the implementation of the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is prescribed by the requested actions of GL 99–02. The use of ASTM D3803–1989 allows for increased accuracy in monitoring the degradation of ESF ventilation systems charcoal over time and is a very accurate and reproducible method for determining the realistic capability of charcoal. ASTM D3803–1989 is considered a more stringent testing standard than the previous referenced standard, ASTM D3803–1986. Additionally, as specified in GL 99–02, a safety factor of 2 has been utilized in the calculation of the revised allowable penetration based upon the credited efficiency approved by the NRC. The proposed change provides assurance for compliance with the current licensing basis regarding dose limits of GDC 19 of Appendix A to 10 CFR 50 and 10 CFR 100. Therefore, the implementation of the proposed change does 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 1, 1999

Description of amendment request: Technical Specification Surveillance Requirement (SR) 3.6.1.7.4 requires that each containment spray nozzle be verified unobstructed on a 10-year frequency. The proposed amendment would revise the frequency for SR 3.6.1.7.4 from once every 10 years to only those conditions when maintenance is performed which could result in nozzle blockage.

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

The proposed change revises the surveillance frequency from every 10 years to following maintenance that could result in nozzle blockage. Analyzed events are initiated by the failure of plant structures, systems or components. The containment spray system is not considered as an initiator of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated, is not significantly increased.

The proposed change revises the Surveillance Frequency. Reduced testing is acceptable where operating experience has shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. The proposed containment spray nozzle Surveillance Frequency has been established based on achieving acceptable levels of equipment reliability. This change does not affect the plant design. Due to the plant design, the spray header is maintained dry and alarmed on water intrusion. Formation of significant corrosion products is unlikely. Due to its location at the top of the containment, introduction of foreign material from exterior to the header is unlikely. Since maintenance that could introduce foreign material is the most likely cause for obstruction, testing or inspection following such maintenance would verify the nozzle(s) being unobstructed, and the system would be capable of performing its safety function. As a result, the consequences of any accident previously evaluated are not significantly affected.

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

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

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

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

The margin of safety for this system is based on the capacity of the spray headers. Since the system is not susceptible to corrosion induced obstruction or obstruction from external to the system, and performance of maintenance on the system would require evaluation of the potential for nozzle blockage and the need for a test or inspection, the spray header nozzles will not become blocked in the event that the safety function is required. Therefore, the capacity of the system would remain unaffected. Hence, this change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: November 17, 1999.

Description of amendment request: The proposed amendments for St. Lucie, Units 1 and 2, will revise the current 72-hour action completion allowed outage time (AOT) specified in Technical Specification (TS) 3.8.1.1, Action ''b,'' to allow 14 days to restore an inoperable emergency diesel generator set to operable status. The proposed AOT is based on an integrated review and assessment of plant operations, deterministic design basis factors, and an evaluation of overall plant risk using probabilistic safety assessment techniques.

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 facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments for St. Lucie Unit 1 and Unit 2 will extend the action completion/allowed outage time (AOT) for a single inoperable Emergency Diesel Generator (EDG) from 72 hours to 14 days. The EDGs are designed as backup AC power sources for essential safety systems in the event of a loss of offsite power. As such, the EDGs are not accident initiators, and an extended AOT to restore operability of an inoperable diesel generator would not significantly increase the probability of occurrence of accidents previously analyzed.

The proposed technical specification revisions involve the AOT for a single inoperable EDG, and do not change the conditions, operating configuration, or minimum amount of operating equipment assumed in the plant safety analyses for accident mitigation. Plant defense-in-depth capabilities will be maintained with the proposed AOT, and the design basis for electric power systems will continue to conform with 10 CFR 50, Appendix A, General Design Criterion 17. In addition, a Probability Safety Assessment (PSA) was performed to quantitatively assess the risk-impact of the proposed amendment for each unit. The impact on the early radiological release probability for design basis events was also evaluated and it is concluded that the risk contribution from this proposed AOT is small and consistent with regulatory risk-assessment acceptance guidelines. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments will not change the physical plant or the modes of operation defined in either facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendments are designed to improve EDG reliability by providing flexibility in the scheduling and performance of preventive and corrective maintenance activities. The surveillance intervals or the operability requirements are not changed by the proposal; only the AOT for a single inoperable EDG will be extended. The proposed changes do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin, and design defense-in-depth capabilities are maintained. An integrated assessment of the risk impact of extending the AOT for a single inoperable EDG has determined that the risk contribution is small and is within regulatory guidelines for an acceptable TS change. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 23, 1999.

Description of amendment request: The proposed license amendments are submitted in response to Generic Letter (GL) 99–02, Laboratory Testing of Nuclear-Grade Activated Charcoal, which requires that American Society for Testing and Materials (ASTM) D3803–1989 be used for testing both new and used charcoal in engineered safety feature applications. The proposed amendments would modify Technical Specification (TS) 3/4.6.3, EMERGENCY CONTAINMENT FILTERING SYSTEM, TS 3/4.6.6, POST ACCIDENT CONTAINMENT VENT SYSTEM, and TS 3/4.7.5, CONTROL ROOM EMERGENCY VENTILATION SYSTEM.

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

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of occurrence of an accident previously evaluated for Turkey Point is not altered by the proposed TS changes because no physical modifications are being made to the plant.

The proposed change requires that new and used charcoal in the plant engineered safety feature (ESF) ventilation systems be tested in accordance with ASTM D3803–1989, at a temperature of 30 °C and a relative humidity of 95%. The use of a new or different test standard to satisfy the charcoal surveillance test requirement does not change the radiological consequences of any previously evaluated accident. The adoption of the ASTM standard will, however, require that future charcoal samples from the emergency containment filters be tested for methyl iodide removal rather than elemental iodine removal as permitted by previous test protocols. The revised test method will provide a more uniform test program for the ESF filters, and will not adversely affect the filters affinity for elemental iodine removal. The adoption of the ASTM standard for laboratory analysis of the ESF charcoal does not impact the design bases of the ESF systems, alter post-accident source terms, or modify the removal efficiencies credited in the facility dose calculations.

The ASTM standard is very stringent and has been shown to provide a more reliable measure of the ability of charcoal to fulfill its intended design function, i.e., to remove radioiodine in any chemical form from the attendant plant gas stream, than previous test protocols. Consequently, the adoption of the ASTM standard for laboratory analysis of the ESF charcoal will ensure that Turkey Point is operated in a manner consistent with the licensing basis of the facility as it relates to the protection of the public and the control room operators during radiological accidents.

Based on the above, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not create a new or different type of accident for Turkey Point because no physical plant changes are being made, and no compensatory measures are imposed that would create a new failure scenario. The proposed change only imposes a more stringent surveillance requirement for both new and used charcoal in the plant ESF ventilation systems. Since no new failure modes are associated with the proposed changes, the activity does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed license amendment adopts a more stringent standard for performing laboratory surveillance tests on both new and used charcoal in the ESF ventilation systems. Given the increased accuracy of the proposed test standard, the amendment also supports the adoption of revised acceptance criteria having a lower safety factor to the plant safety analysis limits. The composite change does not impact the design bases of the ESF systems, alter post-accident source terms, or modify the removal efficiencies credited in the facility dose calculations

The margin of safety associated with operation of the ESF ventilation systems is established by the facility dose calculations and the acceptance criteria for system performance defined in 10 CFR 100 and Criterion 19 of Appendix A to 10 CFR 50. The proposed amendments will not change this acceptance criteria nor the calculated dose limits used to establish the current plant-licensing basis.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Corriea.

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

Date of application for amendment: October 12, 1999.

Brief description of amendment: The proposed amendment would revise the Appendix B Environmental Protection Plan of the Crystal River Unit 3 (CR–3) Operating License. The changes incorporate requirements from a biological opinion (BO) issued by the National Marine Fisheries Service (NMFS). The BO reviews the effects of the cooling water intake system on species of sea turtles protected by the Endangered Species Act (ESA). Additionally, other administrative changes are proposed to Appendix B.

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

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

The proposed changes to the CR–3 EPP are administrative in nature and reflect the information provided in the NMFS BO. These changes do not affect the initial conditions, assumptions, or conclusions of the CR–3 accident analyses. In addition, the proposed changes do not affect the operation or performance of any equipment assumed in the accident analyses. Therefore, the proposed changes would not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature and reflect information provided by the NMFS BO regarding the incidental taking of species of sea turtles protected by the ESA. These changes do not impact or alter the configuration or operation of the facilities and do not create any new modes of operation. Therefore, the proposed changes would not create the possibility of a new or different kind of accident.

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

As indicated above, the proposed changes do not change the configuration or operation of the plant and do not affect the CR–3 accident analyses. The proposed changes are administrative in nature and do not affect any margin of safety for CR–3. Therefore, the proposed changes would not 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: R. Alexander Glenn, General Counsel (MAC–BT15A), Florida Power Corporation, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard Correia.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: October 29, 1999.

Description of amendment request: The proposed license amendment would modify the Technical Specifications (TSs) to: (1) Add operating limits for make-up tank (MUT) level and pressure in a new figure 3.3.1; (2) add surveillance requirements for the MUT pressure instrument channel; (3) change the frequency of calibration for the MUT level instrument from F (every 24 months) to R (refueling interval); (4) change the frequency of calibration for the high pressure injection (HPI) and low pressure injection (LPI) flow instruments; and (5) make minor 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. The proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

The changes included in this LCA [License Change Application] impose new requirements for MU/HPI system operation and testing and extension of calibration frequencies for the MUT level, HPI flow and LPI flow instruments. These changes could not result in initiation of any accident previously evaluated. Therefore, the probability of an accident could not be affected by changes to the MU/HPI system.

As described in the list of benefits for operation with the MU/HPI cross-connect valves open, listed in Section III.B above [Section III.B of the October 29, 1999 application], the purpose of changing the operation of the MU/HPI system was to preclude the possibility of HPI pump damage. The addition of surveillance requirements for the MUT pressure instrument and the addition of LCO [limiting conditions for operation] limits on MUT level and pressure along with an appropriate action statement and AOT [allowed outage time] will ensure that gas entrainment of the MUT does not occur. The proposed change in instrument calibration frequencies will continue to maintain the required accuracy of the MUT level, HPI flow, and LPI flow instruments.

Minor editorial changes are included in this request to improve clarity and readability of the T.S. and could not adversely affect plant operation.

Therefore, the proposed changes will not adversely impact the reliability of the MU/HPI system and could not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

This LCA does not involve the addition of any new hardware. Along with minor editorial changes, the requested changes involve MU/HPI system operation and testing, which could only affect RCS [reactor coolant system] coolant inventory changes during operation and the ability to provide protection in the event of a Loss of Coolant Accident (LOCA). The full spectrum of LOCAs has been evaluated in the FSAR [Final Safety Analysis Report]. Therefore, no new accident scenarios have been created.

The additional controls on MUT level and pressure provided by this LCA will ensure that a malfunction of a different type, gas entrainment of the MU/HPI pumps, will not occur. These limits on MUT level and pressure ensure that the initial conditions assumed for ECCS [emergency core cooling system] operation are maintained. The T.S. limits maintain the accident analysis initial conditions such that no operator action is required to meet NPSH [net positive suction head] or to avoid gas entrainment during ECCS operation with the postulated single failure as required by the TMI–1 licensing basis (Reference 14) [of the October 29, 1999, application].

Extension of the calibration frequencies for the HPI level, HPI flow, and LPI flow will continue to maintain the accuracy of these instruments and could not create the potential for any new accident that has not been evaluated.

Minor editorial changes are included in this request to improve the clarity and readability of the T.S. and could not adversely affect plant operation.

Therefore, these changes do not create the potential for any accident different from those that have been evaluated.

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

This LCA includes changes to the MU/HPI system operation and testing and an extension of the calibration frequency for certain instrument[s]. The requested changes will serve to maintain the proper system initial conditions to ensure the ability of the MU/HPI system to provide protection in the event of a Loss of Coolant Accident (LOCA) and maintain the required instrument accuracy for the instruments where changes to a refueling interval frequency are being requested. NRC guidance for addressing the effect on increased surveillance intervals on instrument drift and safety analysis assumptions presented in GL [generic letter] 91–04 has been addressed in enclosure 1A above [of the October 29, 1999, application].

Minor editorial changes are included in this request to improve the clarity and readability of the T.S. and could not adversely affect plant operation.

These changes, which are consistent with the TMI–1 licensing and design basis requirements, do not result in a degradation of safety related equipment, and therefore, do not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: November 17, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) values for methyl iodide penetration for the main control room environmental control system and the standby gas treatment system. Also, editorial revisions are being made to portions of TS Section 5.0 to reference the correct sections of Regulatory Guide 1.52.

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

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

This proposed revision makes changes to Technical Specification (TS) Section 5.5.7, ''Ventilation Filter Testing Program'' (VFTP). The references to sections in the Regulatory Guide 1.52, Revision 2 for VFTP are being corrected. Additionally, the proposed revision also changes the allowable methyl iodide penetration percent for the carbon in the Standby Gas Treatment (SGT) and the Main Control Room Environmental Control (MCREC) systems when tested in accordance with ASTM DS3803–1989. This is based on the values that would be derived using a factor of safety of 2 between the credited and tested carbon efficiencies. This safety factor is contained in the Generic Letter 99–02. The Generic Letter allows the reduction of the factor of safety between the credited and tested carbon efficiencies from 5 (for systems with heaters) and 7 (for systems without heaters) to 2 (for systems with or without heaters) when tested per ASTM D–3803–1989. Since the factor of safety of 2 is maintained, this change does not involve a significant increase in the probability or the consequences of a previously evaluated event. The changes in the section references to Regulatory Guide 1.52 Revision 2 for the Ventilation Filter Testing Program (VFTP) are considered to be editorial corrections.

2. The change does not involve a significant increase in the probability of or the consequences of an event not previously analyzed.

This proposed revision makes changes to TS Section 5.5.7, ''Ventilation Filter Testing Program'' (VFTP). The section references to Regulatory Guide 1.52 Revision 2 for the Ventilation Filter Testing Program (VFTP) are being corrected. The change in the allowable methyl iodide penetration percent is based on the values that would be derived using the safety factor of 2 contained in Generic Letter 99–02. The Generic Letter will reduce the factor of safety between the credited and tested carbon efficiencies from 5 (for systems with heaters) and 7 (for systems without heaters) to 2 if tested per ASTM D–3803–1989. Since the credited carbon efficiencies in the dose calculations are not being compromised, this change will not involve a significant increase in the probability of, or the consequences of an event not previously analyzed.

The changes in the section references to Reg. Guide 1.52 are editorial and thus do not significantly increase the probability of, or the consequences of a previously unanalyzed event.

3. The change does not significantly reduce the margin of safety.

The change in the allowable methyl iodide penetration percent implements the Generic Letter's carbon efficiency safety factor of 2 between the credited and the tested carbon efficiencies. Per the generic letter, it is acceptable to use this new safety factor since the new standard is more accurate and demanding than previous ones. Therefore, the proposed revision will not significantly reduce the margin of safety. The changes in the section references for Regulatory Guide 1.52 Revision 2 are considered to be editorial corrections.

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

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

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: November 15, 1999 (TS 99–016).

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) for Watts Bar Unit 1 to: (1) revise the Watts Bar TS and associated TS Bases for TS 3.6.11.5 to change the methodology and frequency for sampling the ice condenser ice bed (stored ice) and (2) add a new TS 3.6.11.7 and associated TS Bases to address sampling requirements for all ice additions to the ice bed.

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

A. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a main steam line break (MSLB) inside containment. However, the ice condenser is not postulated as being the initiator of any LOCA or MSLB. This is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the reactor coolant or main steam systems. Since the proposed changes to the TS and TS Bases are solely to revise and provide clarification of the ice sampling and chemical analysis requirements, and are not the result of or require any physical change to the ice condenser, then there can be no change in the probability of an accident previously evaluated in the Safety Analysis Report (SAR).

In order for the consequences of any previously evaluated event to be changed, there would have to be a change in the ice condenser's physical operation during a LOCA or MSLB, or in the chemical composition of the stored ice. The proposed changes do not alter either from existing requirements, except to add an upper limit on boron concentration, which is the bounding value for the Hot Leg Switchover timing calculation. Though the frequency of the existing surveillance requirement for sampling the stored ice is changed from once every 18 months to once every 54 months, the sampling requirements are strengthened overall with (1) the requirement to obtain one randomly selected sample from each ice condenser bay (24 total samples) rather than nine ''representative'' samples, and (2) the addition of a new surveillance requirement to verify each addition of ice meets the existing requirements for boron concentration and pH value. The only other change is to clarify that each sample of stored ice is individually analyzed for boron concentration and pH, but that the acceptance criteria for each parameter is based on the average values obtained for the 24 samples. This is consistent with the bases for the boron concentration of the ice, which is to ensure the accident analysis assumptions for containment sump pH and boron concentration are not altered following complete melting of the ice condenser. Historically, chemical analysis of the stored ice has had a very limited number of instances where an individual sample did not meet the boron or pH requirements, with all subsequent evaluations (follow up sampling) showing the ice condenser as a whole was well within these requirements. Requiring chemical analysis of each sample is provided to preclude the practice of melting all samples together before performing the analysis, and to ensure the licensee is alerted to any localized anomalies for investigation and resolution without the burden of entering a 24 hour ACTION Condition, provided the averaged results are acceptable. Thus, based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The Proposed Change Does Not Create The Possibility Of A New Or Different Kind Of Accident From Any Accident Previously Evaluated.

Because the TS and TS Bases changes do not involve any physical changes to the ice condenser, any physical or chemical changes to the ice contained therein, or make any changes in the operational or maintenance aspects of the ice condenser as required by the Tech Specs, there can be no new accidents created from those already identified and evaluated.

C. The Proposed Change Does Not Involve A Significant Reduction In A Margin Of Safety.

The ice condenser Technical Specifications ensure that during a LOCA or SLB the ice condenser will initially pass sufficient air and steam mass to preclude over pressurizing lower containment, that it will absorb sufficient heat energy initially and over a prescribed time period to assist in precluding containment vessel failure, and that it will not alter the bulk containment sump pH and boron concentration assumed in the accident analysis. Since the proposed changes do not physically alter the ice condenser, but rather only serve to strengthen and clarify ice sampling and analysis requirements, the only area of potential concern is the effect these changes could have on bulk containment sump pH and boron concentration following ice melt. However, this is not affected because there is no change in the existing requirements for pH and boron concentration, except to add an upper limit on boron concentration. This upper limit is the bounding value for the Hot Leg Switchover timing calculation. Averaging the pH and boron values obtained from analysis of the individual samples taken is not a new practice, just one that was not consistently used by all ice condenser plants. Using the averaged values provides an equivalent bulk value for the ice condenser, which is consistent with the accident analysis for the bulk pH and boron concentration of the containment sump following ice melt. Changing the performance frequency for sampling the stored ice does not reduce any margin of safety because (1) the newly proposed surveillance (SR 3.6.15.7) ensures ice additions meet the existing boron concentration and pH requirements, (2) there are no normal operating mechanisms, including sublimation, that reduce the ice condenser bulk pH and boron concentration, and (3) the number of required samples has been increased from nine to 24 (one randomly selected ice basket per bay), which is approximately the same number of samples that would have been taken in the same time period under the existing requirements. Thus, it can be concluded that the proposed TS and TS Bases changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard Correia.

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: November 20, 1998 and July 19, 1999 (TS99–014).

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear plant Unit 1 Technical Specifications (TS) and associated TS Bases to alter the acceptance criteria in Surveillance Requirement (SR) 3.6.11.4 and to revise the Bases for TS 3.6.12. The changes would replace the current visual inspection requirement that uses a 0.38 inch ice/frost buildup criterion with a visual surveillance program that provides an increased confidence level that flow blockage in ice condenser baskets does not exceed the 15 percent assumed in the accident analyses. The proposed amendment dated July 19, 1999 is considered to supercede and replace entirely a proposed amendment dated November 20, 1998 on this same subject.

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

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

Neither the TS amendment nor the TS Bases changes can increase the probability of occurrence of any analyzed accident because they are not the result or cause of any physical modification to ice condenser structures, and for the current design of the ice condenser, there is no correlation between any credible failure of it and the initiation of any previously analyzed event.

Regarding the consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the containment subcompartment and steel containment vessel pressures immediately following the initiation of a LOCA [loss-of-coolant accident] or HELB [high energy line break]. Conservative subcompartment pressure analysis shows this criteria will be met if the reduction in the flow area per bay provided for ice condenser air/steam flow channels is less than or equal to 15 percent, or if the total flow area blocked within each lumped analysis section is less than or equal to the 15 percent assumed in the safety analysis. The present 0.38 inch frost/ice buildup surveillance criteria only addresses the acceptability of any given flow channel, and has no direct correlation between flow channels exceeding this criteria and percent of total flow channel blockage. In fact, it was never the intent of the current SR to make such a correlation. If problems were encountered in meeting the 0.38 inch criteria, it was expected that additional inspection and analysis, such as provided in the proposed amendment, would be performed to make such a determination.

Verifying an ice bed is left with less than or equal to 15 percent flow channel blockage at the conclusion of a refueling outage assures the ice bed will remain in an acceptable condition for the duration of the operating cycle. During the operating cycle, a certain amount of ice sublimates and reforms as frost on the colder surfaces in the Ice Condenser. However, frost does not degrade flow channel area. The surveillance will effectively demonstrate operability for an allowed 18 month surveillance period. Therefore, limiting ice bed flow channel blockage to less than or equal to 15 percent ensures operation is consistent with the assumptions of the design basis accident (DBA) analyses. Thus, the proposed amendment for flow blockage determination provides the necessary assurance that flow channel requirements are met without additional evaluations, and thus will not increase the consequences of a LOCA or HELB.

In regard to the TS 3.6.12 Bases change, clarifying that Condition B does not apply when personnel are standing on or opening doors for a short duration to perform surveillances or minor maintenance activities, such as ice removal, does not increase analyzed accident consequences. These are not new or additional actions to those performed previously, the probability of an accident versus the time to perform these actions is small, the number of personnel involved is small, and their duration is generally much less than the four hour frequency of Required Action B.1 (monitor maximum ice condenser temperature). Therefore, these activities do not adversely affect ice bed sublimation, melting, or ice condenser flow paths. However, if during these activities any door is determined to be restrained, not fully closed from a previous activity, or otherwise not operable, then separate entry into Condition B is required for each door so identified.

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

For such a possibility to exist, there would have to be either a physical change to the ice condenser, or some change in how it is operated or physically maintained. None of the above is true for the proposed TS amendment and TS Bases change.

There is no change to the existing design requirements or inputs/results of any accident analysis calculations.

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

Design Basis Accident analyses have shown that with 85 percent of the total flow area available (uniformly distributed), the ice condenser will perform its intended function. Thus, the safety limit for ice condenser operability is a maximum 15 percent blockage of flow channels. SR 3.6.11.4 currently uses a specific value of 0.38 inch buildup to determine if unacceptable frost/ice blockage exists in the ice condenser. However, this specific value does not have a direct correlation to the safety limit for blockage of ice condenser flow area. The proposed TS amendment requires more extensive visual inspection (33 percent of the flow area/bay) than is currently described (2 flow channels/bay) in the TS Bases for SR 3.6.11.4, thus providing greater reliability and a direct relationship to the analytical safety limits. Changing the TS to implement a surveillance program that is more reliable and uses acceptance criteria of less than or equal to 15 percent flow blockage, as allowed by the TMD [transient mass distribution] analysis, will not reduce the margin of safety of any TS.

Additionally, verifying an ice bed is left with less than or equal to 15 percent flow channel blockage at the conclusion of a refueling outage assures the ice bed will remain in an acceptable condition for the duration of the operating cycle. During the operating cycle, a certain amount of ice sublimates and reforms as frost on the colder surfaces in the Ice Condenser. However, frost has been determined to not degrade flow channel flow area. Thus, design limits for the continued safe function of containment subcompartment walls and the steel containment vessel are not exceeded due to this change.

The change made to TS 3.6.12 Bases does not affect the margin of safety as defined in any TS as it does not involve design specifications or acceptance criteria. This change only adds a clarifying note that entry into Condition B is not required solely because of actions (standing on and opening intermediate/upper deck doors) necessary for the performance of required ice condenser surveillances, maintenance, or routine activities. This does not preclude entry into Condition B during performance of these activities should an intermediate deck door or upper deck door otherwise be determined inoperable.

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

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

NRC Section Chief: Richard P. Correia.

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

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

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

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

Date of amendment request: November 8, 1999.

Description of amendment request: The amendment changed action statements, definitions, and footnotes pertaining to the Technical Specifications for primary containment leakage and primary containment purge system to allow an alternative approach to the existing requirement.

Date of publication of individual notice in **Federal Register**: November 16, 1999 (64 FR 62228).

Expiration date of individual notice: December 16, 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).

CBS Corporation, Docket No. 50–22, Westinghouse Test Reactor, Waltz Mill, Pennsylvania

Date of application for amendment: September 7, 1999, as supplemented on October 1, 1999.

Brief description of amendment: This amendment reassigns the responsibilities of the Site Manager, who works for the Westinghouse Electric Company (a contractor to CBS), to the TR–2 Decommissioning Project Director, who works for CBS.

Date of issuance: November 23, 1999.

Effective Date: November 23, 1999.

Amendment No: 10.

Facility License No. TR–2: This amendment changes the decommissioning plan.

Date of initial notice in **Federal Register**: October 20, 1999 (64 FR 56529).

The Commission has issued a Safety Evaluation for this amendment dated November 23, 1999.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket No. 50–254, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois

Date of application for amendment: March 30, 1999.

Brief description of amendment: The amendment revises the Technical Specifications by changing Surveillance Requirement 4.6.E.2 to allow a one-time extension of the 18-month requirement to pressure set test or replace one half of the Main Steam Safety Valves to an interval of 24 months.

Date of issuance: November 30, 1999.

Effective date: Immediately, to be implemented within 60 days.

Amendment No.: 191.

Facility Operating License No. DPR–29: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* May 5, 1999 (64 FR 24194).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 6, 1999.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to expand the allowable values for Interlocks P–6 (Intermediate Range Neutron Flux) and P–10 (Power Range Neutron Flux) in TS 3.3.1, Table 3.3.1–1, Function 16, Reactor Trip System Interlocks, as recommended by Westinghouse.

Date of issuance: November 30, 1999.

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

Amendment Nos.: Unit 1–189; Unit 2–170.

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

Date of initial notice in **Federal Register***:* May 19, 1999 (64 FR 27319).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 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: May 5, 1999.

Brief description of amendment: This amendment conforms the license to reflect the transfer of Operating License NPF–58 for the Perry Nuclear Power Plant, Unit 1, to the extent held by Duquesne Light Company, to the Cleveland Electric Illuminating Company as previously approved by an Order dated September 30, 1999.

Date of issuance: December 3, 1999.

Effective date: December 3, 1999.

Amendment No.: 108.

Facility Operating License No. NPF–58: This amendment revised the operating license.

Date of initial notice in **Federal Register***:* June 14, 1999 (64 FR 31879).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 27, 1999, as supplemented October 4, 1999.

Brief description of amendments: Revises the Technical Specifications (TS) to extend the allowed outage time, on a one-time basis, for an inoperable emergency diesel generator from 72 hours to 7 days, to replace the Unit 3 diesel engine radiators prior to April 2000. The revision applies to Turkey Point Unit 3 only, however, Unit 4 is included administratively because the TS are combined for both Units.

Date of issuance: November 19, 1999.

Effective date: As of date of issuance, to be implemented prior to April 2000.

Amendment Nos.: 202 and 196.

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

Date of initial notice in **Federal Register***:* August 25, 1999 (64 FR 46441). The supplemental letter of October 4, 1999, provided clarification information that did not change the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 8, 1998.

Brief description of amendments: The proposed amendments would change the Technical Specifications for both units to place tighter restrictions on the allowed outage time for the refueling water storage tank water level instrumentation.

Date of issuance: November 30, 1999.

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

Amendment Nos.: 232 and 215.

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

Date of initial notice in **Federal Register***:* August 31, 1999 (64 FR 47532). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 10, 1999.

Brief description of amendments: The amendments revise Technical Specification (TS) 3/4.4.7 so that the surveillance requirement does not need to be performed when the reactor is defueled with no forced circulation. The revision to TS 3/4.4.7 also includes changes to Tables 3.4–1 and 4.4–3. TS Table 4.4–3 is revised to change the reactor coolant system (RCS) chemistry sampling frequency from three times per 7 days with a maximum interval of 72 hours to a frequency of at least once per 72 hours. An editorial change to Unit 1 Tables 3.4–1 and 4.4–3 relocates the asterisk for the footnote to a position adjacent to the parameter ''dissolved oxygen,'' from its current position next to the allowable chemistry limit in Table 3.4–1 and the analysis frequency in Table 4.4–3. An editorial change also corrects the footnote for Table 3.4–1 for Unit 1 and Unit 2 by making the word ''limit'' plural, as it applies to both the steady-state and transient limits. Surveillance Requirement 4.11.2.2 is revised to delete the phrase ''by analysis of the Reactor Coolant System noble gases.''

Date of issuance: November 19, 1999.

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

Amendment Nos.: 231 and 214.

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

Date of initial notice in **Federal Register***:* October 6, 1999 (64 FR 54376).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 1999.

No significant hazards consideration comments received: No.

PECO Energy Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: June 7, 1999.

Brief description of amendment: The amendment revised the technical specifications (TSs) to reflect the permanent deactivation in the closed position of the ''wet'' instrument reference leg isolation valve HV–61–102. Specifically, TS Table 3.6.3.1, ''Primary Containment Isolation Valve,'' and its associated notations were revised to reflect this current plant configuration.

Date of issuance: November 18, 1999.

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

Amendment No.: 138.

Facility Operating License No. NPF–39. This amendment revised the TSs.

Date of initial notice in **Federal Register**: October 6, 1999 (64 FR 54380).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 15, 1999, as supplemented January 18 and October 22, 1999.

Brief description of amendment: The amendment provides a revision to the Technical Specifications for the FitzPatrick Nuclear Power Plant by modifying the description of what constitutes an acceptable Local Power Range Monitor calibration.

Date of issuance: November 22, 1999.

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

Amendment No.: 257.

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

Date of initial notice in **Federal Register***:* April 10, 1999 (64 FR 11965).

The January 18, 1999, and October 22, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 22, 1999.

Brief description of amendment: This amendment changes the Technical Specifications by extending the pressure-temperature (P-T) limit curves to 24 effective full-power years (EFPY) and 32 EFPY. The current P-T limit curves are valid through 16 EFPY.

Date of issuance: November 29, 1999.

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

Amendment No.: 258.

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

Date of initial notice in **Federal Register***:* August 11, 1999 (64 FR 43775).

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

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: June 30 1997, as supplemented by letters of February 22, March 19, June 30, and October 4, 1999.

Brief Description of amendments: The amendments change the Technical Specifications (TS) to clarify surveillance requirements for the control room emergency filtration system, penetration room filtration system, and related storage pool ventilation system. The changes also revised the required number of radiation monitoring instrumentation channels, and deleted the containment purge exhaust filter TS.

Date of issuance: November 23, 1999.

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

Amendment Nos.: 145 and 136.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register***:* September 1, 1999 (64 FR 47870).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 23, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 21, 1999.

Brief description of amendment: The amendment increases the required volume of stored fuel in the diesel fuel oil storage tank as a result of a conservative recalculation of diesel generator fuel consumption.

Date of Issuance: November 22, 1999.

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

Amendment No.: 180.

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

Date of initial notice in Federal Register: October 20, 1999 (64 FR 56537). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 22, 1999.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 21, 1999, as supplemented by letter dated November 5, 1999.

Brief description of amendment: The amendment extended the effective full implementation date by six months, from December 31, 1999, to June 30, 2000, for Amendment No. 120 issued March 22, 1999, that approved a modification to increase the storage capacity of spent fuel assemblies at the site. The extension is due to delays fabricating and installing the new fuel storage racks.

Date of issuance: November 30, 1999.

Effective date: November 30, 1999, to be implemented by June 30, 2000.

Amendment No.: 129.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 20, 1999 (64 FR 56538). The supplemental letter of November 5, 1999, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 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 January 14, 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).

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

Date of application of amendments: November 17, 1999.

Brief description of amendments: The amendments revised the Technical Specifications to modify the definition of steam generator repair limit for axial tube imperfections detected between the primary side surface of the tube sheet clad and the end of the tube.

Date of Issuance: December 3, 1999.

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

Amendment Nos.: Unit 1–308; Unit 2–308; Unit 3–308.

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

Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendments, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on December 2, 1999. The notice was published in the ''Greenville News,'' Greenville, SC; and the ''Anderson Independent-Mail,'' Anderson, SC, on November 24, 1999. No comments have been received.

The Commission's related evaluation of the amendments, finding of exigent circumstances, consultation with the State of South Carolina, and final no significant hazards consideration determination are contained in a Safety Evaluation dated December 3, 1999.

Attorney for licensee: Richard W. Blackburn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: November 10, 1999 (PCN–510).

Brief description of amendments: The amendments modify the Technical Specification Limiting Condition for Operation 3.4.9.b to delete the phrase stating that two groups of pressurizer heaters be ''capable of being powered from an emergency power supply.

Date of issuance: November 22, 1999.

Effective date: November 22, 1999.

Amendment Nos.: Unit 2–161; Unit 3–152.

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

Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendments, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff by close of business November 19 , 1999. The notice was published in the ORANGE COUNTY REGISTER on November 15–16, 1999. No public comments were received.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated November 22, 1999.

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

NRC Section Chief: Stephen Dembek.

Dated at Rockville, Maryland, this 8th day of December 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–32311 Filed 12–14–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

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

The draft guide, temporarily identified by its task number, DG–1082 (which should be mentioned in all correspondence concerning this draft guide), is titled ''Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.'' This guide is being developed to propose guidance on implementing certain provisions of the NRC's Maintenance Rule by endorsing a revised Section 11 of an industry guideline, NUMARC 93–01, ''Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' which was prepared by the Nuclear Energy Institute.

This draft guide has not received complete staff approval and does not represent an official NRC staff position.

Comments may be accompanied by relevant information or supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by January 10, 2000.

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415–5905; e-mail *CAG@NRC.GOV.* For information about the draft guide and the related documents, contact Mr. W.E. Scott at (301) 415–1020; e-mail MJD1@NRC.GOV.

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

Regulatory guides are available for inspection at the Commission's Public Document Room, 2120 L Street NW., Washington, DC. Requests for single copies of draft or final guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section; or by fax to (301) 415–2289, or by e-mail to <DISTRIBUTION@NRC.GOV>. Telephone requests cannot be accommodated. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

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

Dated at Rockville, Maryland, this 26th day of November 1999.

For the Nuclear Regulatory Commission.

Charles E. Ader,

Director, Program Management, Policy, Development & Analysis Staff, Office of Nuclear Regulatory Research.

[FR Doc. 99–32488 Filed 12–14–99; 8:45 am]

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