[FR Doc. 03–7737 Filed 3–31–03; 8:45 am]

BILLING CODE 4510–30–C

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice (03–035)]

Notice of Prospective Patent License

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of prospective patent license.

SUMMARY: NASA hereby gives notice that SeatSignal, Incorporated, of Atlanta, Georgia, has applied for an exclusive license to practice the invention described in NASA Case Numbers LAR 16324–1 and LAR 16324–1–PCT entitled ''Self-Activating System And Method For Alerting When An Object Or A Person Is Left Unattended,'' for which a U.S. Patent Application was filed and assigned to the United States of America as represented by the Administrator of the National Aeronautics and Space Administration. Written objections to the prospective grant of a license should be sent to Langley Research Center. NASA has not yet made a determination to grant the requested license and may deny the requested license even if no objections are submitted within the comment period.

DATES: Responses to this notice must be received by April 16, 2003.

FOR FURTHER INFORMATION CONTACT: Kurt G. Hammerle, Patent Attorney, Langley Research Center, Mail Stop 212, Hampton, VA 23681–2199. Voicemail: 757–864–2470, Facsimile: 757–864–9190.

Dated: March 25, 2003.

Robert M. Stephens,

Deputy General Counsel.

[FR Doc. 03–7757 Filed 3–31–03; 8:45 am]

BILLING CODE 7510–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of March 31, April 7, 14, 21, 28, May 5, 2003.

AGENCY: Nuclear Regulatory Commission.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of March 31, 2003

There are no meetings scheduled for the Week of March 31, 2003.

Week of April 7, 2003–Tentative

Friday, April 11, 2003

9 a.m. Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360).

This meeting will be webcast live at the Web address: *http://www.nrc.gov.*

12:30 p.m. Discussion of Management Issues (Closed—Ex. 2)

Week of April 14, 2003—Tentative

There are no meetings scheduled for the Week of April 14, 2003.

Week of April 21, 2003—Tentative

There are no meetings scheduled for the Week of April 21, 2003.

Week of April 28, 2003—Tentative

There are no meetings scheduled for the Week of April 28, 2003.

Week of May 5, 2003—Tentative

There are no meetings scheduled for the Week of May 5, 2003.

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: David Louis Gamberoni, (301) 415–1651.

Additional Information

By a vote of 5–0 on march 25, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Final Rule: part 2, subpart G, rules of General Applicability, 'Availability of Official Records' '' be held on March 27, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: March 27, 2003.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–7921 Filed 3–28–03; 11:43 am]

BILLING CODE 7590–01–M

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 March 7, 2003 through March 20, 2003. The last biweekly notice was published on March 18, 2003 (68 FR 12946).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By May 1, 2003, 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,[1] which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* 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: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.*

[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text to 10 CFR 2.714(d), please *see* 67 FR 20884; April 29, 2002.

A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: May 10, 2002, as supplemented March 12, 2003.

Description of amendment request: Carolina Power & Light Company (the licensee) is proposing changes to Appendix A, Technical Specifications (TS), and appendix B, Additional Conditions, of Facility Operating License No. DPR–23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). These changes will revise the licensing basis for HBRSEP2 to implement the Alternative Source Term (AST) described in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' Implementation of the AST will allow for removal of the cycle operating length restriction from appendix B, Additional Conditions, of the Operating License, as the AST radiological consequence analyses support operation for an entire cycle at the increased power level approved in License Amendment No. 196. The AST is used by the licensee in evaluating the radiological consequences of the following Updated Final Safety Analysis Report Chapter 15 accidents:

• Main Steam Line Break,
• Reactor Coolant Pump Shaft Seizure,
• Single Rod Control Cluster Assembly Withdrawal,
• Steam Generator Tube Rupture,
• Large Break Loss-of-Coolant Accident, and
• Waste Gas Decay Tank Rupture.

In addition, revised atmospheric dispersion factors for onsite and offsite dose consequences have been calculated and incorporated in the reanalysis of these events.

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

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

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

Implementation of the Alternative Source Term does not affect the design or operation of HBRSEP, Unit No. 2. Rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences of the postulated accident. The implementation of the Alternative Source Term has been evaluated in revisions to limiting design basis accidents at HBRSEP, Unit No. 2. Based on the results of these analyses, it has been demonstrated that, with the requested changes to the Technical Specifications, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC. This guidance is presented in 10 CFR 50.67 and Regulatory Guide 1.183. The proposed Technical Specifications changes result in more restrictive requirements and support the revisions to the limiting design basis accident analyses.

Therefore, this 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 Previously Evaluated.

The proposed changes do not affect plant structures, systems or components. The Alternative Source Term and those plant systems affected by implementing the proposed changes do not initiate design basis accidents.

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 Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes are associated with the implementation of a new licensing basis for HBRSEP, Unit No. 2. The new licensing basis implements an Alternative Source Term in accordance with 10 CFR 50.67 and the associated Regulatory Guide 1.183. The results of the revised limiting design basis analyses are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies in accordance with the regulatory guidance. The dose consequences of the limiting design basis events are within the acceptance criteria found in the regulatory guidance associated with Alternative Source Terms.

The proposed changes continue to ensure that doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Specifically, the margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are conservatively set below the 10 CFR 50.67 limits. With respect to control room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limits defined by 10 CFR 50, Appendix A, [General Design] Criterion 19 (GDC–19)) continues to be satisfied.

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

Based on the above discussion, Progress Energy Carolinas, Inc., also known as Carolina Power and Light Company, has determined that the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602–1551.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: February 17, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specification Surveillance Requirement 3.10.1.9 to require that the Standby Shutdown Facility (SSF) diesel generator (DG) be loaded to at least 3280 kilowatts during the surveillance. The current requirement is that the SSF DG be loaded to at least 3000 kilowatts during the surveillance. The change supports resolution of an Oconee design basis issue associated with SSF pressurizer heater capacity.

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:

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC in 10 CFR 50.92. This ensures 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:

This change revises the loading of the Standby Shutdown Facility (SSF) Diesel Generators (DG) to ≥ 3280 kW. The design rating of the DG is currently 3500 kW. Since the proposed loading is within the design rating already evaluated, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As stated above, the proposed revision revises the DG loading to an analytical value that is within the equipment's design limit. Applicable load and support system calculations have been revised and results have shown that the increase does not adversely affect the ability of the SSF diesel generator or SSF to perform its intended safety function. Additionally, this change is bounded by all of the existing accidents and does not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. Therefore, the proposed change does not involve a reduction in a margin of safety.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: January 31, 2003.

Description of amendment request: The proposed amendments would revise Appendix A, Technical Specifications (TS) 3.4.11, ''RCS Pressure and Temperature (P/T) Limits,'' to incorporate revised P/T curves. The revised P/T curves are based on calculations performed in accordance with General Electric (GE) Topical Report NEDC–32983P, ''General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation.'' The NEDC–32983P methodology is consistent with the guidance contained in Regulatory Guide (RG) 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' dated March 2001.

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

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

The proposed changes request for LaSalle County Station, Units 1 and 2, that the pressure and temperature (P/T) limit curves in TS 3.4.11, ''RCS Pressure and Temperature (P/T) Limits,'' and Surveillance Requirement (SR) 3.4.11.1 and SR 3.4.11.2 be revised. The revised curves were developed using the methodology of GE Topical Report NEDC–32983P, ''General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation.'' NEDC–32983P methodology has been previously approved by the NRC for use by licensees. The P/T limits 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 reactor coolant pressure boundary, a condition that is unanalyzed. Thus, the proposed changes do not have any affect on the probability of an accident previously evaluated.

The P/T curves are used as operational limits during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. The P/T curves provide assurance that station operation is consistent with previously evaluated accidents. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes do not change the response of plant equipment to transient conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

The proposed changes adopt P/T curves that have been developed using the methodology of GE Topical Report NEDC–32983P. The NEDC–32983P methodology is consistent with the guidance contained in RG 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' dated March 2001. In a letter dated September 14, 2001, the NRC approved NEDC–32983P for use by licensees.

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

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief : Anthony J. Mendiola.

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

Date of amendment request: February 27, 2003.

Description of amendment request: The proposed amendments revise the Technical Specifications to reflect a one-time deferral of the primary containment Type A leak rate test to no later than July 22, 2009, for Unit 1 and no later than May 16, 2008, for Unit 2.

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

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

Response: No.

The proposed change will revise Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, Technical Specification (TS) 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than July 22, 2009, for Unit 1, and no later than May 16, 2008, for Unit 2. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor coolant system (RCS) following a design basis loss of coolant accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Therefore, extending this test interval on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of Type A testing provides assurance that the QCNPS primary containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

Response: No.

The proposed change for a one-time extension of the Type A tests for QCNPS, Units 1 and 2, will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

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

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

Response: No.

QCNPS, Units 1 and 2, are General Electric BWR/3 [boiling water reactor class 3] plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the RCS; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests do not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, EGC [Exelon Generation Company, LLC] does not expect additional degradation, during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: February 27, 2003.

Description of amendment request: The proposed amendments add a surveillance requirement to perform a quarterly trip unit calibration of the reactor protection system scram discharge volume water level—high differential pressure switches.

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

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

Response: No.

The proposed Technical Specifications (TS) change adds a trip unit calibration surveillance requirement (SR) for the analog trip units associated with the Scram Discharge Volume (SDV) Water Level—High Trip Function for the Reactor Protection System (RPS) Instrumentation. Specifically, SR 3.3.1.1.11 is added to Function 7.b of TS Table 3.3.1.1–1, ''Reactor Protection System Instrumentation.'' In addition, the proposed change revises Function 7.a of TS Table 3.3.1.1–1 to delete a reference to thermal switches, applicable to Unit 1 through cycle 17. The change to Function 7.a is editorial, since Unit 1 SDV level instrumentation has been upgraded to replace Fluid Components International thermal switches with Magnetrol float switches.

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 proposed changes will not involve an increase in the probability of an accident previously evaluated. Additionally, these proposed changes do not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. The proposed changes establish requirements that ensure 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.

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

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

Response: No.

There is no change being made to the parameters within which Quad Cities Nuclear Power Station (QCNPS) is operated. The proposed changes do not adversely impact the manner in which the SDV Water Level—High RPS instrumentation will operate under normal and abnormal operating conditions. The proposed changes will not alter the function demands on credited equipment. No alteration in the procedures, which ensure QCNPS remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. Therefore, these proposed 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 proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms and actions. The proposed changes do not affect the probability of failure or availability of the affected instrumentation, and the proposed changes do not revise any allowable values for RPS functions. Therefore, it is concluded that the proposed changes do 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: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

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: January 14, 2003.

Description of amendment request: This license amendment request proposes a change to Technical Specifications (TSs) 5.1.1, 5.4.1, and 5.5.1 that would replace the requirement for the plant manager to approve administrative procedures and the Offsite Dose Calculation Manual. The plant manager approval signature would be replaced with the signature of a procedurally authorized individual who would be a more appropriate authority for approval of the activity.

Basis for proposed no significant hazards consideration determination: As required by Section 50.91(a) of Title 10 of the Code of Federal Regulations (10 CFR), 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 replace the plant manager's approval with the approval by an authorized individual is consistent with the requirements of Regulatory Guide 1.33 and American National Standards Institute (ANSI) N18.7–1976/American Nuclear Society (ANS) 3.2. The authorized individuals are management and supervisory personnel who satisfy the requirements of ANSI N18.1–1971. Use of ANSI N18.1–1971 is consistent with the requirements of the existing TS and Updated Safety Analysis Report (USAR). The change is administrative and does not impact or otherwise affect the physical plant.

The proposed change to the License Condition to delete the reporting time frame eliminates duplication of a requirement that is already an integral part of 10 CFR 50.73 which is referenced in the License Condition. The proposed change is administrative and does not impact or otherwise affect the physical plant.

Therefore, the proposed 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 or different kind of accident from any previously evaluated. The proposed administrative changes do not involve any physical modifications to the facility nor add new equipment. The methods of plant operation have not been altered. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative in nature and have no direct impact upon any plant safety analyses. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: 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 (PNPP), Unit 1, Lake County, Ohio

Date of amendment request: January 30, 2003.

Description of amendment request: This license amendment request would modify the existing minimum critical power ratio (MCPR) safety limit contained in Technical Specification (TS) 2.1.1.2. Specifically, the change modifies the MCPR safety limit values, as calculated by Global Nuclear Fuel (GNF), by decreasing the limit for two recirculation loop operation from 1.10 to 1.07, and decreasing the limit for single recirculation loop operation from 1.11 to 1.08. The change resulted from the core reload analysis performed for the Perry Nuclear Power Plant (PNPP) fuel cycle 10.

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.

PPNP Updated Safety Analysis Report (USAR) Section 4.2, "Fuel System Design," states the PNPP fuel system design bases are provided in the General Electric Topical Report, NEDE–24011–P–A, "General Electric Standard Application for Reactor Fuel (GESTAR II)." The MCPR Safety Limit is one of the limits used to protect the fuel in accordance with the design basis. The MCPR Safety Limit establishes a margin to the onset of transition boiling. The basis of the MCPR Safety Limit remains the same, ensuring that greater than 99.9% of all fuel rods in the core avoid transition boiling. The methodology used to determine the MCPR Safety Limit values is contained within GESTAR II and is NRC approved. The change does not result in any physical plant modifications or physically affect any plant components. As a result, there is no increase in the probability of occurrence of a previously analyzed accident.

The fundamental sequences of accidents and transients have not been altered. The Safety Limit MCPR is established to avoid fuel damage in response to anticipated operational occurrences. Compliance with a MCPR Safety Limit greater than or equal to the calculated value will ensure that less than 0.1% of the fuel rods will experience boiling transition. This in turn ensures fuel damage does not occur following transitions due to excessive thermal stresses on the fuel cladding. The MCPR Operating Limits are set higher (*i.e.*, more conservative) than the Safety Limit such that potentially limiting plant transients prevent the MCPR from decreasing below the MCPR Safety Limit during the transient. Therefore, there is no impact on any limiting USAR Appendix 15B transients. The radiological consequences remain the same as previously stated in the USAR. Therefore, the consequences of an accident do not increase over previous evaluations in the USAR.

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

The MCPR Safety Limit basis is preserved, which is to ensure that transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the postulated limiting transient. The values are calculated in accordance with GESTAR II. The GESTAR II analyses have been accepted by the NRC.

The MCPR Safety Limit is one of the limits established to ensure the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. No changes in the design of the plant or the method of operating the plant are associated with these revised safety limit values. Therefore, no new or different kind of accident from any previously evaluated is created.

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

This change revises the PNPP MCPR Safety Limit values. The new MCPR Safety Limit values reflect changes due to the Cycle 10 core reload, but do not alter the design or function of any plant system, including the fuel. The new MCPR Safety Limit values were calculated using NRC-approved methods described in GESTAR II. The proposed MCPR Safety Limit values continue to satisfy the fuel design safety criteria which ensures that transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the postulated limiting transient. Therefore, the proposed values for the MCPR Safety Limit 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: December 10, 2002.

Description of amendment request: The proposed amendment would revise the Unit 2 reactor coolant system (RCS) pressure-temperature curves in Technical Specification (TS) Figures 3.4–2 and 3.4–3 and associated TS Bases. The revised curves will bound operation of the unit for the remainder of its current license duration and bound operation with planned license amendments to increase the power level at which the unit is allowed to operate. In support of this proposed amendment, Indiana Michigan Power (I&M) has submitted a request, in accordance with 10 CFR 50.60, ''Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,'' for exemption from requirements in 10 CFR Part 50, Appendix G, ''Fracture Toughness Requirements.''

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated

The proposed change will revise the RCS pressure-temperature curves to reflect new limiting reactor vessel materials, to bound operation of the reactor up to 3600 MWt for the current fuel cycle and beyond, to reflect new fluence analysis methodology, to reflect the use of ASME [American Society of Mechanical Engineers] Code Case N–641, to include boltup limits, and to no longer include instrument uncertainty margins.

The proposed change will not result in physical changes to structures, systems, or components (SSCs), or changes to event initiators or precursors. The proposed change will not affect the ability of personnel to control RCS pressure at low temperatures and, thereby, ensure the integrity of the reactor coolant pressure boundary. Use of Code Case N–641 in developing the proposed revision to the RCS pressure-temperature curves is in accordance with methodologies accepted by the ASME. These methodologies provide assurance that the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes, and provide an acceptable level of protection against brittle failure.

Additionally, the proposed changes will not impact the design or operation of plant systems such that previously analyzed SSCs will be more likely to fail. The initiating conditions and assumptions for accidents described in the UFSAR [updated final safety analysis report] will remain as previously analyzed. Therefore, the proposed changes will not involve a significant increase in the probability of an accident previously evaluated.

Consequences of an Accident Previously Evaluated

The proposed change does not reduce the ability of any SSC to limit the radiological consequences of accidents described in the UFSAR. The proposed change will not alter any assumptions made in the analysis of radiological consequences of previously evaluated accidents, nor does it affect the ability to mitigate these consequences. No new or different radiological source terms will be generated as a result of the proposed change. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The format changes will improve the appearance of the affected pages but will not affect any requirements. In summary, the probability of occurrence and the consequences of an accident previously evaluated will not be significantly increased.

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

Response: No.

The proposed change will not result in physical changes to SSCs. The proposed change will not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor will it alter the design of any plant systems. The proposed change solely involves RCS pressure-temperature limits. The types of potential accidents associated with these limits have been previously identified and evaluated. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or single failures will be introduced as a result of the proposed changes. No new or different modes of failure will be created. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed RCS pressure-temperature curves will continue to provide adequate margins of protection for the reactor coolant pressure boundary. The proposed changes have been determined, through supporting analyses, to be in accordance with the methodologies and criteria set forth in the applicable regulations, or in accordance with technically adequate alternatives. Compliance with these methodologies provides adequate margins of safety and ensures that the reactor coolant pressure boundary will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events as described in the UFSAR. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will 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 requests involve no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: February 25, 2003.

Description of amendment request: The proposed Technical Specification

(TS) changes will add an allowed outage time (AOT) for Engineered Safety Features Actuation System (ESFAS) Instrumentation channels to be out of service in a bypassed state.

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

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

The addition of an ACTION STATEMENT and the addition of an AOT (and its associated actions if not met) for a TS action statement are neither an accident initiator nor precursor. The ESFAS actuates in response to an accident and has a mitigating function. Increasing the TS requirements for specific TS instrument loops provides additional assurance that the channels will be capable of performing their design function in the event of a DBA [design-basis accident]. The ability of the operations staff to respond to an evaluated accident or plant transient will not be hampered. This change provides conservative requirements to assure that the design basis of the plant is maintained.

Addition of conservative changes to the Engineered Safety Feature Actuation System Instrumentation does not contribute to the initiation of any accident evaluated in the FSAR [Final Safety Analysis Report]. Supporting factors are as follows:

• The changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–1431, Rev. 2. There are no deletions from the Technical Specifications made by these changes, nor relaxation in any applicability, action, or surveillance requirements.

• Overall plant performance and operation is not altered by the proposed changes. There are to be no plant hardware changes as a result of this proposed change and only minimal procedural changes.

Therefore, since the Engineered Safety Feature Actuation System Instrumentation are treated more conservatively, the probability of occurrence or consequences of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

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

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

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. Additionally, the addition of an ACTION STATEMENT and an AOT with conservative requirements are intended to assure that the plant is in a safe configuration and can meet accident analyses assumptions. These changes are conservative and consistent with the Improved Technical Specifications, NUREG–1431, Rev. 2. No new accident initiator mechanisms are introduced since:

• No physical changes to the Engineered Safety Feature Actuation System Instrumentation are made.

• No deletions from the Technical Specifications are made.

• No relaxations in any applicability, action, or surveillance requirements are made.

Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed, which determine safe plant operation.

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

The proposed change requires that an instrument channel for an Engineered Safety Feature remain operable or be restored to operability within a reasonable time period, otherwise a controlled shutdown is required. This conforms to the safety analysis where the plant and its systems, structures and components must be capable of performing the safety function while a DBA is occurring, in the presence of a worst case single failure.

This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–0452, Rev. 5. The proposed changes impose more restrictive operating limitations, and their use provides increased assurance that the Engineered Safety Feature Actuation System Instrumentation remains operable. Since the changes are conservative additions, it is concluded that the changes do not involve a significant reduction in the margin of safety. This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

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 G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 13, 2003.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 4.2.1, Fuel Assemblies, to modify the fuel design description to encompass Framatome Advanced Nuclear Power (FANP) fuel assemblies and also to modifiy TS 4.3, Fuel Storage, to remove nomenclature specific to Global Nuclear Fuels analysis methods. The proposed TS changes are needed to allow the receipt and storage of Framatome fuel assemblies.

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 in accordance with the three standards set forth in 10 CFR 50.92(c), which are presented below:

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

No. The proposed amendment revises TS 4.2.1, Fuel Assemblies, to modify the fuel design description to accommodate FANP fuel designs. The change to TS 4.2.1 is administrative and simply adds descriptive text to reflect that FANP fuel assemblies have a water channel.

To make the fuel storage TS compatible with the storage of GNF [Global Nuclear Fuels] and FANP fuel, the proposed amendment also modifies TS 4.3, Fuel Storage, to delete criteria specific to GNF fuel storage criticality analysis methods. BFN criticality analysis and storage requirements continue to be adequately described in the Updated Final Safety Analysis Report (UFSAR) and in existing TS 4.3.1.1.b, TS 4.3.1.1.c, TS 4.3.1.2.b, 4.3.1.2.c, and 4.3.1.2.d. Hence, the proposed elimination of the GNF-specific criteria in TS 4.3 does not affect BFN design basis requirements associated with ensuring adequate criticality margins are maintained for fuel storage.

The requested TS changes do not involve any plant modifications or operational changes that could affect system reliability, performance, or the possibility of operator error. The requested changes do not affect any postulated accident precursors, do not affect accident mitigation systems, and do not introduce any new accident initiation methods. Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes to TS do not affect the performance of any BFN structure, system, or component credited with mitigating any accident previously evaluated. Fuel storage criticality analyses will continue to be performed in accordance with established UFSAR commitments that are independent are fuel vendor specific methods. The TS changes do not introduce new modes of operation or involve plant modifications.

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

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

No. The proposed amendment modifies TS 4.3, Fuel Storage, to remove nomenclature specific to GNF criticality analysis methods. Fuel storage criticality analyses will continue to be performed in accordance with UFSAR commitments and the remaining TS commitments in accordance with FANP accepted methods, which specify appropriate criteria and conservatisms. Therefore, the proposed TS 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 19, 2002.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) Chapter 5.0, ''Administrative Controls,'' to incorporate three approved TS Task Force (TSTF) changes: TSTF–258, Revision 4; TSTF–299, Revision 0; and TSTF–308, Revision 1. These changes have been incorporated in Revision 2 of NUREG 1431, ''Standard Technical Specifications Westinghouse Plants.''

In addition, the amendment proposes two editorial changes. These changes either update personnel titles with the titles currently used at WBN and TVA's other nuclear units or clarify required staffing levels.

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 in accordance with the three standards set forth in 10 CFR 50.92(c), which are presented below:

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

No. The proposed changes affect only administrative requirements or programs. As indicated below, the justification for five of the changes (Parts 2 through 4 of Change Number 2 and Change Numbers 3, 5 [only Parts 1 and 2 of Change 5], 6, and 7) is based on the existence of a regulation or other regulatory document which controls the administrative requirements. For these changes, the proposed amendment modifies the administrative TS to make it consistent with the current regulations or NRC guidance document. Two changes (Change Number 1 and Part 1 of Change Number 2) are strictly editorial. In addition, two changes (Change Number 4 and Part 3 of Change Number 5) add a requirement to make the program consistent with the criteria for Surveillance Requirements in the Improved Standard Technical Specifications (ISTS). Based on the preceding information, the proposed amendment does not involve technical changes to the configuration or operation of the plant there is not a significant increase in the probability or consequences of an accident previously evaluated:

Change No.	Administrative section affected	Justification for the change
1.	5.1, ''Responsibility,'' Section 5.1.2	Editorial update of staff titles.
2.	5.2.2, ''Unit Staff''	Part 1 of Change number 2—Editorial clarification of the number of non-licensed operators required for the operation of WBN Unit 1. Parts 2 through 4 of Change Number 2—The existing administrative requirements are revised to align the requirements with 10 CFR 50.54.
3.	5.3, ''Unit Staff Qualifications,'' Section 5.3.2.	Adds TS 5.3.2 which clarifies the ''Operator'' and ''Senior Operator'' definitions in 10 CFR 55.4 and ties these positions to the requirements of 10 CFR 50.54.
4.	5.7.2.4, ''Primary Coolant Sources Outside Containment.	WBN TS 5.7.2.4 serves the same function as a Surveillance Requirement (SR). The proposed change structures TS 5.7.2.4 so that it is consistent with other ISTS SRs and the frequency extension allowed by SR 3.0.2.
5.	5.7.2.7, ''Radioactive Effluent Controls Program''.	The intent of the revisions to this TS are to: 1) eliminate possible confusion or improper implementation of the requirements of 10 CFR 20; 2) clarifies the wording to not require dose projections for a calendar quarter and a calendar year every 31 days; 3) structures the TS so that it is consistent with other ISTS SRs.
6.	5.9.4, ''Monthly Operating Reports''	The proposed change makes the TS reporting requirements consistent with the reporting requirements in Generic Letter 97–02.
7.	5.11, ''High Radiation Area''	The proposed revision updates the TS to be consistent with 10 CFR 20.1601(c) and updates the acceptable alternate controls to those given in 10 CFR 20.1601.

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

No. As indicated above, the proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods controlling normal plant operation. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed changes will not reduce the margin of safety because they have no effect on assumptions made in WBN's safety analysis or the configuration of plant equipment important to safety. Additionally, several of the proposed revisions adjust the administrative requirements to be consistent with existing regulations or NRC guidance documents and therefore, will not adversely impact plant safety. The balance of the proposed changes are editorial updates or adjust a program to be consistent with the ISTS.

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: Allen G. Howe.

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

Date of amendment request: February 14, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for Watts Bar Nuclear Plant (WBN), Unit 1. The proposed TS change would allow WBN Unit 1 to be refueled and operated using the Westinghouse 17x17 Robust Fuel Assembly-2 (RFA–2) design commencing with Cycle 6 in September 2003.

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

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

No. The Loss of Coolant Accident (LOCA) and non-LOCA transients and accidents which are potentially affected by the parameters and assumptions associated with the use of RFA–2 (including the effects of Tritium Producing Burnable Absorber Rods, TPBARs) have been evaluated/analyzed and all design standards and applicable safety criteria are met. The consideration of these changes does not result in a situation where the design material, and construction standards that were applicable prior to the change are altered. Therefore, the changes occurring with the use of RFA–2 will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The changes associated with the use of RFA–2 do not affect plant systems such that their function in the control of radiological consequences is adversely affected. TVA's evaluation documents that the design standards and applicable safety criteria limits continue to be met and, therefore, fission barrier integrity is not challenged. The fuel rod design (the first fission product barrier) is not changed. Compared to the current grid design on the resident fuel, the RFA–2 grid design provides improved resistance to fuel rod fretting. The RFA–2 fuel changes have been shown not to adversely affect the response of the plant to postulated accident scenarios. These changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Final Safety Analysis Report (FSAR).

Therefore, since the actual plant configuration, performance of systems, and initiating event mechanisms are not being changed as a result of this evaluation, TVA has concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with the use of RFA–2 do not result in a change to the design basis of any plant component or system. The evaluation of the effects of the use of RFA–2 shows that all design standards and applicable safety criteria limits are met. Specifically, the results of the evaluations/analyses lead to the following conclusions:

1. The RFA–2 fuel design for Watts Bar Unit 1 is mechanically compatible with the current fuel assemblies, core components, the control rods and the reactor internals interfaces.

2. The structural integrity of the RFA–2 fuel design has been evaluated for seismic/LOCA loadings for Watts Bar Unit 1. Evaluation of the RFA–2 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the fuel assembly design is structurally acceptable.

3. The changes to the nuclear characteristics due to the transition to the RFA–2 fuel assembly design will be within the range normally seen from cycle to cycle due to fuel management.

4. The RFA–2 fuel assembly design is hydraulically compatible with the current fuel assemblies.

5. The core design and safety analyses documented in this report demonstrate the capability of the core to operate safely at the rated Watts Bar Unit 1 design thermal power with either a mixed core of RFA–2 fuel and the current fuel product or with a full core of RFA–2 fuel.

6. TVA's amendment request establishes a reference upon which to base Westinghouse reload safety evaluations for future reloads with the RFA–2 fuel assembly design.

7. Reload core designs with either a mixed core of RFA–2 fuel and the current fuel product or with a full core of RFA–2 fuel are compatible with the planned introduction of Tritium-Producing Burnable Absorber Rods (TPBARs) into Watts Bar Unit 1.

These changes therefore do not cause the initiation of any accident nor create any new failure mechanisms. All equipment important to safety will operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The use of RFA–2 is not expected to result in more adverse conditions and is not expected to result in any increase in the challenges to safety systems.

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

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

No. The margin of safety is maintained by assuring compliance with acceptance limits reviewed and approved by the NRC. All of the appropriate acceptance criteria for the various analyses and evaluations have been met, therefore, there has not been a reduction in any margin of safety.

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

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

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

NRC Section Chief: Allen G. Howe.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action *see* (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide

Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: April 10, 2002.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to relocate emergency diesel generator maintenance inspection requirements from Section 4.7 to the Updated Final Safety Analysis Report.

Date of Issuance: March 7, 2003.

Effective date: March 7, 2003 shall be implemented within 30 days of issuance, except the relocation of the emergency diesel generator maintenance requirements of Technical Specification 4.7, which shall be incorporated into the Updated Final Safety Analysis Report in accordance with the schedule specified by 10 CFR 50.71.

Amendment No.: 236.

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

Date of initial notice in ***Federal Register****:* May 28, 2002 (67 FR 36926).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 30, 2002, as supplemented November 21 and December 16, 2002, and January 23, 2003.

Brief description of amendment: This amendment revises the Technical Specifications by eliminating the requirements to perform response time testing for several reactor protection system and engineered safety feature functions in conformance with previously approved topical reports.

Date of issuance: March 7, 2003.

Effective date: March 7, 2003.

Amendment No.: 112.

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

Date of initial notice in ***Federal Register****:* November 1, 2002 (67 FR 61676).

The November 21 and December 16, 2002, and January 23, 2003, letters provided clarifying information and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 28, 2002.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.9.9, ''Containment Ventilation Isolation System,'' to allow the same administrative controls for this TS as were approved previously by the NRC in Amendment No. 104 to the HNP TS for TS 3/4.9.4, ''Containment Building Penetrations,'' to provide consistency between the two TS.

Date of issuance: March 12, 2003.

Effective date: March 12, 2003.

Amendment No.: 113.

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

Date of initial notice in ***Federal Register****:* October 1, 2002 (67 FR 61676).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2003.

No significant hazards consideration comments received: No.

Duke Energy Corporation, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of application for amendments: October 10, 2002, as supplemented by letters dated February 7 and February 26, 2003.

Brief description of amendments: The amendment authorizes the licensee to continue to use, for operational cycle 13 beginning in March 2003, and subsequent cycles of operation, the reactor coolant system cold leg elbow tap flow coefficients that were approved by the NRC on an interim basis for cycle 12 in Amendment No. 186.

Date of issuance: March 19, 2003.

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

Amendment Nos.: 199.

Facility Operating License No. NPF–52: Amendment authorizes revision of the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register****:* November 26, 2002 (67 FR 70765).

The supplements dated February 7 and February 26, 2003, provided clarifying information that did not change the scope of the October 10, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 29, 2002, as supplemented by letters dated September 25 and November 12, 2002, and January 8 and January 29, 2003.

Brief description of amendments: The amendments revised the Technical Specifications to allow a one-time change in the Appendix J, Type A containment integrated leakage rate test interval from the currently required 10-year interval to a test interval of 15 years.

Date of issuance: March 12, 2003.

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

Amendment Nos.: 205/198.

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

Date of initial notice in ***Federal Register****:* July 9, 2002 (67 FR 45563).

The supplements dated September 25 and November 12, 2002, and January 8 and January 29, 2003, provided clarifying information that did not change the scope of the May 29, 2002, application or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 12, 2003.

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: May 29, 2002, as supplemented by letters dated September 25 and

November 12, 2002, and January 8 and January 29, 2003.

Brief description of amendments: The amendments revised the Technical Specifications to allow a one-time change in the Appendix J, Type A containment integrated leakage rate test interval from the currently required 10-year interval to a test interval of 15 years.

Date of issuance: March 12, 2003.

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

Amendment Nos.: 211/192.

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

*Date of initial notice in **Federal Register**:* July 9, 2002 (67 FR 45563).

The supplements dated September 25 and November 12, 2002, and January 8 and January 29, 2003, provided clarifying information that did not change the scope of the May 29, 2002, application or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 12, 2003.

No significant hazards consideration comments received: No.

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: April 24, 2002, as supplemented by letters dated July 18, December 18 and 20, 2002, and February 19, 2003.

Brief description of amendment: The amendment reflects a full-scope implementation of the alternative source term, as described in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' pursuant to 10 CFR 50.67, ''Accident source term.''

Date of issuance: March 14, 2003.

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

Amendment No.: 132.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* June 11, 2002 (67 FR 40021).

The July 18, December 18 and 20, 2002, and February 19, 2003, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2003.

No significant hazards consideration comments received: No.

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: May 14, 2002, as supplemented by letters dated February 12 and 28, 2003.

Brief description of amendment: The amendment modifies the surveillance requirements (SRs) pertaining to the testing of the Division 3 standby emergency diesel generator (EDG). The change allows performance of some required surveillance tests for the Division 3 EDG during any mode of plant operation (previously allowed only in Modes 4 (Cold Shutdown) and 5 (Refueling)).

Date of issuance: March 14, 2003.

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

Amendment No.: 133.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications and Surveillance Requirements.

*Date of initial notice in **Federal Register**:* June 25, 2002 (67 FR 42824).

The February 12, 2003, supplemental letter provided clarifying information and the February 28, 2003, supplemental letter withdrew the requested change to the Note associated with SR 3.8.1.8. The supplemental letters did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 2003.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: June 5, 2002, as supplemented on January 9 and March 4, 2003.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to implement the alternate source term methodology for the fuel-handling accident analysis. Specifically, the amendment revises TS 3.9.3, ''Containment Penetrations,'' to: (1) Permit the equipment closure hatch opening and the personnel airlock doors to be capable of being closed during movement of irradiated fuel, (2) allow use of administrative controls for unisolating containment penetrations during movement of irradiated fuel, (3) delete the containment purge and containment pressure relief requirements and associated surveillances with the reactor subcritical for less than 550 hours, and (4) eliminate the TS applicability ''during core alterations.'' In this regard, the amendment adopts TS Task Force (TSTF) Standard TS Change Travelers TSTF–68, ''Containment Personnel Airlock Doors Open During Fuel Movement,'' TSTF–312, ''Administratively Control Containment Penetrations,'' and, in part, TSTF–51, ''Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations.'' The amendment also revises the Applicability Statements for Limiting Condition for Operation (LCO) 3.3.8 for the fuel storage building emergency ventilation system (FSBEVS) actuation instrumentation and LCO 3.7.13 for the FSBEVS to also add the term ''recently'' before ''irradiated fuel assemblies.'' In addition, the LCO Required Action would likewise be modified to add the term ''recently'' to now require the suspension of movement of recently irradiated fuel in the FSB.

Date of issuance: March 17, 2003.

Effective date: March 17, 2003.

Amendment No.: 215.

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

*Date of initial notice in **Federal Register**:* July 9, 2002 (67 FR 45567).

The January 9 and March 4 letters provided clarifying information that did not expand the scope of the proposed amendment or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2003.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: August 16, 2002.

Brief description of amendment: The amendment relocates certain Control Rod Block functions from Technical Specifications 3/4.2.C, ''Control Rod Block Actuation,'' Tables 3.2.C.1, 3.2.C–2, and 4.2.C to the Updated Final Safety Analysis Report.

Date of issuance: March 17, 2003.

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

Amendment No.: 196.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* November 12, 2002 (67 FR 68735).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 24, 2002, as supplemented by letter dated February 4, 2003.

Brief description of amendment: The amendment revises Technical Specifications (TSs) relating to positive reactivity additions while in shutdown modes by clarifying TSs involving positive reactivity additions. In addition, the borated water volume requirements in TS 3.1.2.7 is now presented in ''percent level'' units and an obsolete reference from Surveillance Requirement 4.8.2.2 is deleted.

Date of issuance: March 7, 2003.

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

Amendment No.: 185.

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

*Date of initial notice in **Federal Register**:* December 10, 2002 (67 FR 75874).

The February 4, 2003, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 2, 2001, as supplemented by letters dated September 24, 2001, and February 27, July 31, and December 19, 2002.

Brief description of amendment: The Refueling Water Storage Pool (RWSP) purification system is aligned to the RWSP to maintain the purity and clarity of the borated water contained in the pool. It is also one of two means of makeup to the Spent Fuel Pool, with the Condensate Storage Pool being the primary makeup source. Entergy Operations Inc. has proposed to revise its Waterford Steam Electric Station, Unit 3, Updated Final Safety Analysis Report (UFSAR) to allow the manual valves (FS–423 and FS–404) that isolate the RWSP from the RWSP purification system and provide the boundary between the seismically qualified, safety related RWSP and the non-seismic, non-safety related RWSP purification system to be maintained open.

Date of issuance: March 12, 2003.

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

Amendment No.: 186.

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

*Date of initial notice in **Federal Register**:* May 16, 2001 (66 FR 27176).

The September 24, 2001, and February 27, July 31, and December 19, 2002, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 27, 2002.

Brief description of amendments: These amendments delete technical specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminate the requirements to have and maintain the post accident sampling system at the Dresden Nuclear Power Station, Units 2 and 3. The amendments also address related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: March 11, 2003.

Effective date: As of the date of issuance and shall be implemented within 180 days.

Amendment Nos.: 197/190.

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

*Date of initial notice in **Federal Register**:* January 21, 2003 (68 FR 2802).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: November 21, 2002, as supplemented February 25, 2003.

Brief description of amendment: This amendment revised the Technical Specifications (TSs) for the safety limit for the minimum critical power ratio from its current value of 1.09 to 1.07 for two recirculation-loop operations, and from 1.11 to 1.09 for single recirculation-loop operation.

Date of issuance: March 11, 2003.

Effective date: As of the date of issuance, to be implemented prior to startup for Cycle 8 operations, scheduled for March 2003.

Amendment No.: 127.

Facility Operating License No. NPF–85: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* January 7, 2003 (68 FR 802). The February 25, 2003, letter provided additional information that clarified the application, 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 as published in the **Federal Register** on January 7, 2003 (68 FR 802).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 27, 2002.

Brief description of amendments: These amendments delete technical specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminate the requirements to have and maintain the post accident sampling system at the Quad Cities Nuclear Power Station, Units 1 and 2. The amendments also address related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: March 11, 2003.

Effective date: As of the date of issuance and shall be implemented within 180 days.

Amendment Nos.: 212/206.

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

*Date of initial notice in **Federal Register**:* January 21, 2003 (68 FR 2802)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 2003.

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: March 14, 2002, as supplemented by letters dated July 17 and September 12, 2002, and January 24, 2003.

Brief description of amendment: This amendment supplements License Amendment No. 100, which was issued on February 24, 1999, by placing restrictions on removing the inclined fuel transfer system (IFTS) blind flange during Operational Modes 1, 2, and 3. The amendment includes a time limit on the removal of the IFTS blind flange, provides a requirement to install the upper pool IFTS gate prior to IFTS blind flange removal, and limits the unbolted configuration of the IFTS blind flange when it is rotated.

Date of issuance: March 7, 2003.

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

Amendment No.: 123.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* February 4, 2003 (68 FR 5675).

The supplemental information contained clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 30, 2002.

Brief description of amendment: This amendment deletes Technical Specification (TS) 5.5.3, "Post Accident Sampling System (PASS)," and thereby eliminates the requirements to have and maintain the PASS at the Perry Nuclear Power Plant, Unit 1. The amendment also addresses related changes to TS 5.5.2, "Primary Coolant Sources Outside Containment."

Date of issuance: March 7, 2003.

Effective date: As of the date of issuance and shall be implemented within 180 days.

Amendment No.: 124.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* January 21, 2003 (68 FR 2803).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 14, 2002, supplemented by a letter dated January 24, 2003, that supersedes previous applications dated August 9, 2000, June 13, 2002.

Brief description of amendment request: The amendment revises TS 6.5.4 and 6.5.3 to eliminate the requirements for the Independent Onsite Safety Review Group (IOSRG) which is not needed for safe monitoring of TMI–2 based on consideration that the reactor has been defueled to the extent reasonably achievable and the fuel shipped offsite. The amendment also revises TS 6.4 to delete the requirements for unit staff training that are outdated based on the adoption of a systems approach to training consistent with 10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel."

Date of issuance: March 5, 2003.

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

Amendment No.: 59.

Facility Operating License No. DPR–73: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 6, 2002 (67 FR 50955).

The November 14, 2002, application and supplemental letter dated January 24, 2003, replace in their entirety the previous applications dated August 9, 2000, June 13, 2002. The November 14, 2002, application supplemented by the January 24, 2003, letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 5, 2003.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendment: July 23, 2002, as supplemented November 15, 2002, and January 24, 2003.

Brief description of amendment: The amendment revises the Unit 2 reactor coolant system pressure-temperature curves in Technical Specification (TS) Figures 3.4–2 and 3.4–3 and associated TS Bases. The revised curves will bound operation of the unit for the remainder of its current license duration and bound operation with planned license amendments to increase the power level at which the unit is allowed to operate.

Date of issuance: March 20, 2003.

Effective date: As of the date of issuance and shall be implemented prior to startup from Unit 2 refueling outage 14.

Amendment No.: 255.

Facility Operating License No. DPR–74: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 29, 2002 (67 FR 66010).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 2003.

No significant hazards consideration comments received: No.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: January 13, 2000, and supplemented by letters dated June 1, 2001, August 13, 2001, and October 15, 2002.

Brief description of amendment: The amendment adds License Condition 2.B.(9) to the MY license. This new license condition incorporates the, Nuclear Regulatory Commission (NRC) approved, "License Termination Plan Rev 3." (LTP), and associated addendum, into the MY license and allows the licensee to make certain changes to the approved LTP without prior NRC review and approval.

Date of issuance: February 28, 2003.

Effective date: Date of issuance; to be implemented within [30] days from the date of issuance.

Amendment No.: 168.

Facility Operating License No. DPR–36: The amendment adds License Condition 2.B.(9).

Date of initial notice in ***Federal Register:*** March 19, 2002.

The supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the **Federal Register** on March 19, 2002.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation Report dated February 28, 2003.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: September 26, 2002.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: March 6, 2003.

Effective date: March 6, 2003.

Amendment No.: 197.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75882).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 2003.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: November 15, 2002, as supplemented by letter dated February 24, 2003.

Brief description of amendment: The amendment revises the safety limit minimum critical power ratio values in Technical Specification 2.1.1.2.

Date of issuance: March 17, 2003.

Effective date: March 17, 2003.

Amendment No.: 198.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 24, 2002 (67 FR 78521).

The supplemental letter provided clarifying information that was within the scope of the original **Federal Register** Notice (67 FR 78521) and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 26, 2002, as supplmented December 19, 2002.

Brief description of amendment: The amendment revises Technical Specification (TS) 1.0, ''Definitions,'' TS 2.1, ''Safety Limits, Reactor Core,'' TS 2.3, ''Limiting Safety System Settings, Protective Instrumentation,'' TS 3.1, ''Reactor Coolant System,'' TS 3.8, ''Refueling Operations,'' TS 3.10, ''Control Rod and Power Distribution Limits,'' TS 6.9, ''Reporting Requirements,'' and their associated Bases. These modifications allow the licensee to implement a Core Operating Limits Report (COLR) by relocating cycle-specific, reactor coolant system-related parameter limits from the TSs to the COLR. In addition, the amendment makes administrative changes to the above TSs.

Date of issuance: March 11, 2003.

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

Amendment No.: 165.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 3, 2002 (67 FR 56322).

The supplemental information dated December 19, 2002, contained clarifying information and did not change the scope of the July 26, 2002, application nor the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 19, 2002, supplemented by letters dated September 13 and October 21, 2002.

Brief description of amendment: The amendment revises the current radiological consequence analyses for the Kewaunee Nuclear Power Plant (KNPP) design-basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors'' and Pursuant to 10 CFR 50.67, ''Accident Source Term.''

Date of issuance: March 17, 2003.

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

Amendment No.: 166.

Facility Operating License No. DPR–43: Amendment revised the current radiological consequence analyses for the KNPP design-basis accidents to implement the AST.

Date of initial notice in ***Federal Register:*** April 16, 2002 (67 FR 18646).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2003.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: January 4, 2002, as supplemented January 9, 2003.

Brief description of amendment: The amendment adds a limiting condition for operation of the mechanical vacuum pump instrumentation to trip the pumps on indication of high radiation levels in the main steam line and adds associated Surveillance Requirements.

Date of issuance: March 11, 2003.

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

Amendment No.: 143.

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

Date of initial notice in ***Federal Register:*** February 19, 2002 (67 FR 7421).

The January 9, 2003, supplement contained clarifying information and did not change the staff's proposed finding of no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 2003.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: August 20, 2002.

Brief description of amendment: The amendment modifies the diesel generator action statements and surveillance requirements defined in the plant's Technical Specifications, in order to reduce degradation of the diesel generators associated with fast starting and rapid loading.

Date of issuance: March 17, 2003.

Effective date: March 17, 2003.

Amendment No.: 144.

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

Date of initial notice in ***Federal Register:*** October 1, 2002 (67 FR 61684).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 20, 2002.

Brief description of amendment: The proposed amendment would change Technical Specification (TS) Section 1.10, ''Definitions, Dose Equivalent I–131,'' to allow the use of the thyroid dose conversion factors listed in the International Commission on Radiological Protection Publication No. 30 (ICRP–30), ''Limits for Intakes of Radionuclides by Workers,'' 1979, in determining the iodine-131 dose equivalent reactor coolant activity in TS Section 3/4.4.8 and in calculating the radiological consequences from postulated design basis accidents.

Date of issuance: March 6, 2003.

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

Amendment No.: 162.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 20, 2002 (67 FR 53991).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 2003.

No significant hazards consideration comments received: No.

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 application for amendments: August 9 2002, as supplemented by letters dated January 8 and February 6, 2003.

Brief description of amendments: The amendments revised the Updated Final Safety Analysis Report to incorporate the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance for the surveillance of the material capsules.

Date of issuance: March 10, 2003.

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

Amendment Nos.: 237 and 179.

Renewed Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** October 1, 2002 (67 FR 61684).

The supplements dated January 8 and February 6, 2003, provided clarifying information that did not change the scope of the August 9, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 10, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 24th day of March, 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

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NUCLEAR REGULATORY COMMISSION

Appointments to Performance Review Boards for Senior Executive Service

AGENCY: Nuclear Regulatory Commission.

ACTION: Appointment to Performance Review Boards for Senior Executive Service.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has announced the following appointments to the NRC Performance Review Boards.

The following individuals are appointed as members of the NRC Performance Review Board (PRB) responsible for making recommendations to the appointing and awarding authorities on performance appraisal ratings and performance awards for Senior Executives and Senior Level employees:

Patricia G. Norry, Deputy Executive Director for Management Services, Office of the Executive Director for Operations.
R. William Borchardt, Associate Director for Inspection and Programs, Office of Nuclear Reactor Regulation.
Stephen G. Burns, Deputy General Counsel, Office of the General Counsel.
Frank J. Congel, Director, Office of Enforcement.
James E. Dyer, Regional Administrator, Region III.
Jesse L. Funches, Chief Financial Officer.
Scott F. Newberry, Director, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research.
James B. Schaeffer, Director, Applications Development Division, Office of the Chief Information Officer.
Michael L. Springer, Director, Office of Administration.
Martin J. Virgilio, Director, Office of Nuclear Material Safety and Safeguards.
Michael F. Weber, Deputy Director, Office of Nuclear Security and Incident Response.

The following individuals will serve as members of the NRC PRB Panel that was established to review appraisals and make recommendations to the appointing and awarding authorities for NRC PRB members:

Karen D. Cyr, General Counsel, Office of the General Counsel.
William F. Kane, Deputy Executive Director for Reactor Programs, Office of the Executive Director for Operations.