following procedures apply to public participation in the meeting:

1. Persons who wish to provide a written statement should submit an electronic copy or mail a reproducible copy to Ms. Tull at the contact information listed above. All submittals must be postmarked by August 14, 2007, and must pertain to the topic on the agenda for the meeting.

2. Questions from members of the public will be permitted during the meeting, at the discretion of the Chairman.

3. The transcript and written comments will be available for inspection on NRC's Web site (*www.nrc.gov*) and at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852–2738, telephone (800) 397–4209, on or about November 16, 2007. Minutes of the meeting will be available on or about September 17, 2007.

This meeting will be held in accordance with the Atomic Energy Act of 1954, as amended (primarily Section 161a); the Federal Advisory Committee Act (5 U.S.C. App); and the Commission's regulations in Title 10, *U.S. Code of Federal Regulations,* Part 7.

Dated: July 25, 2007.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E7–14715 Filed 7–30–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Sunshine Act Meetings

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of July 30, August 6, 13, 20, 27, September 3, 2007.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 30, 2007

Thursday, August 2, 2007

1:25 p.m. Affirmation Session (Public Meeting) (Tentative).

a. Dominion Nuclear North Anna, LLC (Early Site Permit for North Anna ESP Site), LBP–07–9 (June 29, 2007) (Tentative).

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

1:30 p.m. Briefing on Risk-Informed, Performance-Based Regulation (Public Meeting) (Contact: John Monninger, 301 415–6189).

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

Week of August 6, 2007—Tentative

There are no meetings scheduled for the Week of August 6, 2007.

Week of August 13, 2007—Tentative

There are no meetings scheduled for the Week of August 13, 2007.

Week of August 20, 2007—Tentative

Tuesday, August 21, 2007

1:30 p.m. Meeting with OAS and CRCPD (Public Meeting) (Contact: Shawn Smith, 301 415–2620).

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

Wednesday, August 22, 2007

9:30 a.m. Periodic Briefing on New Reactor Issues (Morning Session) (Public Meeting) (Contact: Donna Williams, 301 415–1322).

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

1:30 p.m. Periodic Briefing on New Reactor Issues (Afternoon Session) (Public Meeting) (Contact: Donna Williams, 301 415–1322).

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

Week of August 27, 2007—Tentative

There are no meetings scheduled for the Week of August 27, 2007.

Week of September 3, 2007—Tentative

There are no meetings scheduled for the Week of September 3, 2007.

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* 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: Michelle Schroll, (301) 415–1662.

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

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g. braille, large print), please notify the NRC's Disability Program Coordinator, Rohn Brown, at 301–492–2279, TDD: 301–415–2100, or by e-mail at *REB3@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

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: July 26, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–3744 Filed 7–27–07; 12:11 pm]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and 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 July 4, 2007 to July 18, 2007. The last biweekly notice was published on July 17, 2007 (72 FR 39081).

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. Within 60 days after the date of publication of this notice, 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.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place 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, Rulemaking, Directives and Editing 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 O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, 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 2.309, 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 within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include 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/requestor to relief. A petitioner/requestor who fails to satisfy 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.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, 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 by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. 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 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 requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

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 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 PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendment request: June 15, 2007.

Description of amendment request: The amendment proposes to relocate the inservice testing requirements to the administrative section of the technical specifications (TS), remove the inservice inspection activities from TS and locate them in an owner-controlled program, and establish a TS Bases Control Program. All of these changes are proposed to be consistent with NUREG–1431, Revision 3, ''Standard Technical Specifications Westinghouse Plants.''

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (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 amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, and it does not change an accident previously evaluated in the Final Safety Analysis Report (FSAR). The proposed change is administrative in nature, and it will continue to ensure that the inspection and testing requirements required by regulations are met. The American Society of Mechanical Engineers (ASME) Code requirements are established, reviewed and approved by ASME, the industry, and ultimately endorsed by the NRC for inclusion into 10 CFR 50.55a. Updates to the ASME Code reflect advances in technology and consider information obtained from plant operating experience to provide enhanced inspection and testing. Thus, the proposed change will revise TS to appropriately reference the ASME Code required by 10 CFR 50.55a for performing inservice testing, specifically referencing the ASME Code for Operation and Maintenance of Nuclear Power Plants, rather than the ASME Section XI Code.

The proposed change does not affect operations, and the inspection and testing required is not an accident initiator.

Therefore, this amendment 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 of different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated, and it does not change an accident previously evaluated in the Final Safety Analysis Report (FSAR). As noted above, the proposed change is administrative in nature, the inspection and testing required is not an accident initiator, and no new accident precursors are being introduced. The proposed change will revise TS to appropriately reference the ASME Code required by 10 CFR 50.55a for performing inservice testing, which will continue to ensure that the inspection and testing requirements required by regulations are met. Since inservice testing will continue to be performed in accordance with regulations, adequate assurance is provided to ensure that the safety-related pumps and valves will continue to operate as required. No new testing is required that could create a new or different type of accident.

Therefore, this amendment 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?

Response: No.

The proposed amendment does not involve a significant reduction in a margin of safety. The proposed amendment does not adversely affect a plant safety limit or a limiting safety system setting, and does not alter a design basis limit for a parameter evaluated in the FSAR. The proposed change is administrative in nature, and it will continue to ensure that the inspection and testing requirements required by regulations are met. Since inservice testing will continue to be performed in accordance with regulations, adequate assurance is provided to ensure that the safety-related pumps and valves will continue to operate as required and perform their intended safety function.

Therefore, this amendment 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Thomas H. Boyce.

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

Date of amendment request: July 3, 2007.

Description of amendment request: The proposed change relocates the quality and quantity requirements associated with the emergency diesel generator (EDG) fuel oil within the Technical Specifications (TS) through the creation of a new TS Limiting Condition for Operation and the Diesel Fuel Oil Testing Program.

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

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

Response: No.

The proposed changes in the diesel fuel oil testing program will continue to ensure that new and stored diesel fuel oil properties are maintained within specified limits to assure EDG operation. The testing of diesel generator fuel oil is not considered an initiator or a mitigating factor in any previously evaluated accidents.

The deletion of the requirement to drain and inspect the fuel oil storage tank (FOST) does not impact any of the previously analyzed accidents. Periodic testing of the fuel oil as required by the Diesel Fuel Oil Testing Program will identify poor quality oil. Actions are included that will require the quality of the oil to be maintained within acceptable limits. Draining and inspecting the FOST are not considered an accident initiator or mitigating factor in any previously evaluated accidents.

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 results in changes to the existing diesel fuel oil testing program and the deletion of the [Surveillance Requirements] associated with the performance of periodic draining and inspection of the FOSTs. No plant modifications are required to support the proposed TS changes. There is no impact to plant structures, systems, or components, or in the design of the plant structures, systems, or components.

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.

The proposed change does not result in any plant modifications. Diesel generator fuel oil quantity and quality will continue to be maintained within acceptable limits to assure the ability of the EDG to perform its intended function.

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

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

Attorney for licensee: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Exelon Generation Company, LLC, Docket No. 50–237, Dresden Nuclear Power Station (DNPS), Unit 2, Grundy County, Illinois

Date of amendment request: July 10, 2007.

Description of amendment request: The proposed amendment would revise the values of the safety limit minimum critical power ratio (SLMCPR) in Technical Specification (TS) Section 2.1.1, ''Reactor Core SLs.''

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 probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for DNPS, Unit 2, Cycle 21 such that the fuel is protected during normal operation and during plant transients or anticipated operational occurrences (AOOs).

Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during plant transients or AOOs. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation.

The proposed change to the SLMCPR assures that safety criteria are maintained for DNPS, Unit 2, Cycle 21.

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

The SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the current level of fuel protection is maintained by continuing to ensure that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed SLMCPR values were developed using NRC-approved methods. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (*i.e.*, that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) is met.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Russell Gibbs.

Exelon Generation Company, LLC, Docket No. 50–373, LaSalle County Station, Unit 1, LaSalle County, Illinois

Date of amendment request: June 18, 2007.

Description of amendment request: The proposed amendment would revise technical specification TS 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time extension of the LaSalle County Station (LSCS), Unit 1, primary containment Type A Integrated Leak Rate Test (ILRT) date for the current requirement of no later than June 13, 2009, prior to startup following the thirteenth LSCS Unit 1 refueling outage (L1R13).

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?

Response: No

The proposed changes will revise LSCS, Unit 1, TS 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time extension of the primary containment Type A Integrated Leak Rate Test (ILRT) date to ''prior to startup following L1R13.'' The current Type A ILRT interval of 15 years, based on past performance, would be extended on a one-time basis by approximately 5% of the current interval.

The function of the primary containment is to isolate and contain fission products released from the reactor Primary Coolant System (PCS) 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 ILRTs is not a precursor of any accident previously evaluated. Type A ILRTs provide assurance that the LSCS Unit 1 primary containment 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 changes 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 changes do 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?

Response: No

The proposed changes for a one-time extension of the Type A ILRT for LSCS Unit 1 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident 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. Does the change involve a significant reduction in a margin of safety?

Response: No

LSCS Unit 1 is a General Electric BWR/5 plant with a Mark II primary containment. The Mark II primary containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The drywell floor separates the drywell and 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 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 changes for a one-time extension of the Type A ILRT does not affect the method for Type A, B, or C testing or the test acceptance criteria.

EGC has conducted a risk assessment to determine the impact of a change to the LSCS Unit 1 Type A ILRT schedule from a baseline ILRT frequency of three times in ten years to once in 15.67 years (i.e., 15 years plus 8 months) for the risk measures of Large Early Release Frequency (i.e., LERF), Total Population Dose, and Conditional Containment Failure Probability (i.e., CCFP). This assessment indicated that the proposed LSCS ILRT interval extension has a minimal impact on public risk.

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. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Russell Gibbs.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: November 27, 2006.

Description of amendment request: The proposed amendments would modify various technical specification (TS) requirements for emergency diesel generators (EDGs). Specifically, the licensee stated that the proposed changes would eliminate several accelerated tests and a test table, modify acceptance criteria for fast start and load rejection tests, and also, eliminate the EDG failure report. The proposed changes are consistent with the Nuclear Regulatory Commission's (NRC's) regulatory guidance presented in Generic Letter 93–05, ''Line-Item Technical Specifications Improvement to Reduce Surveillance Requirements for Testing During Power Operation,'' Generic Letter 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' and NUREG–1433, Rev. 3.1, ''Standard Technical Specifications, General Electric Plants, BWR/4.''

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 changes are associated with the testing and reporting requirements of the eight (four on each unit) Emergency Diesel Generators (EDGs). The changes will eliminate unnecessary EDG testing requirements that contribute to potential mechanical degradation of the EDGs. The changes are based on the NRC guidance and recommendations provided in Generic Letter 93–05 or Generic Letter 94–01, or are consistent with NUREG–1433. The change to the reporting requirement is administrative in nature.

The probability of an accident is not increased by these changes because the EDGs are not assumed to be initiators of any design basis event. Additionally, the proposed changes do not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, or controlled. The consequences of an accident will not be increased because the changes to the EDGs and associated support systems still provide a high degree of assurance that their operability is maintained.

Therefore, 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.

The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions, associated with the operation of the plant. Accordingly, the proposed changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure or system in the performance of their safety function.

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

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

Response: No.

The proposed changes to the EDGs either: (1) Modify the test acceptance criteria, (2) modify the accelerated testing schedules, or (3) eliminate a reporting requirement. The change to the test acceptance criteria is based on the recommendations of Regulatory Guide 1.9, and the change to the reporting requirement is enveloped by other NRC reporting requirements. The other changes are consistent with NRC guidance, and reduce unnecessary testing and improve EDG reliability. Requirements to assure that a common mode failure has not affected the remaining operable EDGs have been maintained. The existing routine testing frequency, unaffected by these changes, has been shown to be adequate for assuring the EDGs are operable based on operating experience. The proposed changes do not impact the assumptions of any design basis accident, and do not alter assumptions relative to the mitigation of an accident or transient event.

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

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

Attorney for licensee: Mr. Bradley Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Harold K. Chernoff.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 29, 2007.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 Technical Specifications to increase the power level required for a reactor trip following a turbine trip (P–9 setpoint).

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

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

The analysis of the proposed change included an evaluation of loss of load/turbine trip transient. With systems functioning as designed, the proposed change to the P–9 setpoint does not impact [the] accident analyses previously evaluated in the Updated Final Safety Analysis Report (UFSAR). In the best estimate case (normal plant conditions; all control systems functioning per design), the pressurizer power operated relief valves (PORV) and the steam generator safety valves are not challenged following the turbine trip without reactor trip. Consequently, the proposed change does not adversely affect the probability of a small break loss of coolant accident due to a stuck-open PORV. The sensitivity study that assessed the affects of degraded control systems found that a failure of all condenser steam dump valves resulted in challenging the PORVs and the steam generator (SG) safety valves. However, overfilling of the pressurizer will not occur and this Condition 2 event will not initiate a Condition 3 event. The challenge to the PORVs with all steam dump banks failed does not violate design or licensing criteria. Therefore, the proposed setpoint change does not significantly increase 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 setpoint change does not create the possibility of a new or different kind of accident than any accident previously evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change. The proposed Technical Specification changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. The revised setpoint for the P–9 function ensures that accident/transient analyses acceptance criteria continue to be met. This change makes no modifications to the plant that would introduce new accident causal mechanisms and has no affect on how the trip functions operate upon actuation. 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 the margin of safety.

The proposed Technical Specification changes do not involve a significant reduction in a margin of safety. The analyses supporting the proposed change to the P–9 setpoint demonstrate that margin exists between the setpoint and the corresponding safety analysis limits. The calculations are based on plant instrumentation and calibration/

functional test methods and include allowances associated with the setpoint change. The results of analyses and evaluations supporting the proposed change demonstrate acceptance criteria continue to be met. The reactor trip on turbine trip provides additional protection and conservatism beyond that required for protection of public health and safety; the safety analyses in chapter 15 of the UFSAR do not take credit for this reactor trip. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: June 27, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 3.3.3, ''Post Accident Monitoring (PAM) Instrumentation,'' to include containment recirculation sump level instrumentation which will be used for indication of recirculation sump strainer blockage. Additionally, the amendment would revise TS 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' by replacing the term ''trash racks and screens'' with the more descriptive term ''strainers.'' Finally, the amendment would revise TS 3.6.14, ''Containment Recirculation Drains,'' to include Limiting Conditions for Operation, Actions, and Surveillance Requirements to ensure the operability of flow paths credited in the evaluation of potential adverse effects of post-accident debris on the containment recirculation function pursuant to NRC Generic Letter 2004–02.

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.

The proposed change consists of a revision to the Technical Specifications (TS) for post accident monitoring (PAM) instrumentation to include new containment recirculation sump level instrumentation, a revision to the TS for Emergency Core cooling System (ECCS) to replace the term ''trash rack and screen'' with the term ''strainer,'' and a revision to the TS for containment recirculation drains to add two flow paths credited in the evaluation of the effects of post-accident debris on the containment recirculation functions pursuant to Nuclear Regulatory Commission Generic Letter 2004–02.

The proposed TS revisions will not increase the probability of an accident because the associated components, i.e., the new sump level instruments, the new strainers, and the two flow paths, are not, and will not become, accident initiators. The activities involving these components pursuant to the proposed TS revisions consist of implementing Surveillance Requirements for the new sump level instruments and flow paths and actions to be taken if these components are inoperable. These activities will not increase the likelihood of an accident. The TS change associated with the sump strainers is editorial in that it reflects the terminology that has been applied to new pocket strainers that continue to perform the trash rack and screen functions. The change in terminology will not result in any new activities.

The proposed TS revision will not increase the consequences of an accident because the associated components all provide mitigative functions for an accident, and their ability to perform their mitigative functions is not reduced by the associated TS changes. The TS changes associated with the new sump level instrumentation and the recirculation [flow paths] will provide increased assurance that these components will be available to perform their mitigative function if needed. The TS change associated with the sump strainers is editorial and does not affect the mitigative capability of the screens.

Therefore, the proposed change will 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 TS revisions will not create the possibility of a new or different kind of accident from any accident previously evaluated because the associated components, i.e., the new sump level instruments, the new strainers, and the two flow paths, are components that will not initiate any accident. The proposed TS changes associated with these components will not cause them to be operated in any manner not previously evaluated for the specific components or for similar components, or cause them to become other than passive components.

Therefore, the proposed change will 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.

The margin of safety associated with the proposed TS revisions involves the ability of the associated components, i.e., the new sump level instruments, the new strainers, and the two flow paths, to assure the ECCS and containment spray recirculation function can be adequately accomplished. The TS changes associated with the new sump level instrumentation and the recirculation [flow paths] will provide increased assurance that this function can be fulfilled. The TS change associated with the sump strainers is editorial and does not affect this function.

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

The Nuclear Regulatory Commission (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: Kimberly Harshaw, Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Travis Tate.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: May 31, 2007.

Description of amendment request: The proposed amendment would revise the accident source term used in the NMP2 design basis radiological consequence analyses in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.67. The revised accident source term replaces the current methodology that is based on TID–14844, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' with the alternative source term (AST) methodology described in Regulatory Guide (RG) 1.183, ''Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' The amendment request is for full implementation of the AST as described in RG 1.183, with the exception that TID–14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access. Proposed changes include the following: Revision of the Technical Specification (TS) definition of Dose Equivalent I–131 to be consistent with the AST analyses; TS changes that reflect revised design requirements regarding the use of the standby liquid control system (SLCS) to buffer the suppression pool pH to prevent iodine re-evolution following a postulated design basis loss-of-coolant accident (LOCA); revisions to the TS operability requirements for the control room envelope filtration system and the control room envelope air conditioning system, consistent with the assumptions contained in the AST fuel-handling accident (FHA) analysis; and credit for operation of the residual heat removal system in the drywell spray mode for the post-LOCA removal of airborne elemental iodine and particulates from the drywell atmosphere. Because NMPNS is considering an extended power uprate (EPU) project that would increase the maximum licensed reactor core power level to 3,988 megawatts thermal (MWt), the AST analyses have been performed using a bounding core isotopic inventory that is based on operation at 3,988 MWt in lieu of the currently licensed power of 3,467 MWt.

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.

Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs [design-basis accidents]. The AST does not affect the design or manner in which the facility is operated; rather, for postulated accidents, the AST is an input to calculations that evaluate the radiological consequences. The AST does not by itself affect the post-accident plant response or the actual pathway of the radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Implementation of the AST has been incorporated in the analyses for the limiting DBAs at NMP2.

The structures, systems and components affected by the proposed change mitigate the consequences of accidents after the accident has been initiated. Application of the AST does result in changes to NMP2 Updated Safety Analysis Report (USAR) functions (e.g., Standby Liquid Control system [SLCS]). As a condition of application of AST, NMPNS is proposing to use the [SLCS] to control the suppression pool pH following a LOCA. These changes do not require any physical modifications to the plant. As a result, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the NMP2 USAR. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

Plant-specific AST radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the acceptance criteria provided by the NRC for use with the AST. These criteria are presented in 10 CFR 50.67 and Regulatory Guide 1.183. Even though the AST dose limits are not directly comparable to the previously specified whole body and thyroid dose guidelines of General Design Criterion 19 and 10 CFR 100.11, the results of the AST analyses have demonstrated that the 10 CFR 50.67 limits are satisfied. Therefore, it is concluded that adoption of the AST does not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Implementation of AST and the proposed changes does not alter or involve any design basis accident initiators. These changes do not involve any physical changes to the plant and do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

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

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

Response: No.

The changes proposed are associated with a new licensing basis for analysis of NMP2 DBAs. Approval of the licensing basis change from the original source term to the AST is being requested. The results of the accident analyses performed in support of the proposed changes are subject to revised acceptance criteria. The limiting DBAs have been analyzed using conservative methodologies, in accordance with the guidance contained in Regulatory Guide 1.183, to ensure that analyzed events are bounding and that safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67 and Regulatory Guide 1.183. Thus, the proposed changes continue to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as in the control room, are within corresponding regulatory criteria.

Therefore, by meeting the applicable regulatory criteria for AST, it is concluded that 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal.

Pacific Gas and Electric Co., Docket No. 50–133, Humboldt Bay Power Plant (HBPP), Unit 3 Humboldt County, California

Date of amendment request: April 4, 2007.

Description of amendment request: The licensee has proposed amending the existing license to allow the results of near-term surveys, performed on a portion of the plant site, to be included in the eventual Final Status Survey (FSS) for license termination.

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?

Response: No.

The proposed change would allow survey results for a specific area within the licensed site area, performed prior to Humboldt Bay Power Plant (HBPP) Unit 3 decommissioning and dismantlement activities, to be used in the overall licensed site area Final Status Survey (FSS) for license termination. The FSS will be performed following completion of HBPP Unit 3 decommissioning and dismantlement activities. This proposed change would not change plant systems or accident analysis, and as such, would not affect initiators of analyzed events or assumed mitigation of accidents. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration to the plant or require existing equipment to be operated in a manner different from the present design. Implementation of a cross contamination prevention and monitoring plan will be done in accordance with plant procedures and licensing bases documents. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident evaluated.

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

Response: No.

The proposed change has no effect on existing plant equipment, operating practices, or safety analysis assumptions. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Jennifer K. Post, Pacific Gas and Electric Company, 77 Beale Street, B30A, San Francisco, CA.

NRC Branch Chief: Bruce Watson.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: March 22, 2007.

Description of amendment request: The proposed amendment supports full-scope implementation of an alternative source term (AST) methodology, in accordance with Section 50.67, ''Accident source term,'' of Title 10 of the *Code of Federal Regulations* (10 CFR) with the exception that Technical Information Document (TID) 14844, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' will continue to be used as the radiation dose basis for equipment qualification.

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

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

The implementation of AST assumptions has been evaluated in revisions to the analyses of the following limiting DBAs [design-basis accidents].

- Loss-of-Coolant Accident.
- Fuel Handling Accident.
- Control Rod Ejection Accident.
- Locked Rotor Accident.
- Main Steam Line Break Accident.
- Steam Generator Tube Rupture Accident.

Based upon the results of these analyses and evaluations, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events satisfies the dose limits in 10 CFR 50.67 and are within the regulatory guidance provided by the NRC for use with the AST methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not affect the plant response or the actual pathway of the activity released from the fuel. Therefore, it is concluded that AST does not involve a significant increase in the consequences of an accident previously evaluated.

Implementation of AST provides for elimination of the Fuel Handling Building ventilation system filtration TS [Technical Specification] requirements and elimination of Control Room ventilation filtration TS requirements in Modes 5 or 6. It also eliminates containment integrity TS requirements while handling irradiated fuel and during core alterations. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The affected systems are not accident initiators; and application of the AST methodology is not an initiator of a design basis accident.

Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Modes 5 or 6 does not increase the probability of an accident because the proposed change does not affect the design and operational controls to prevent dilution events. These same design and operational controls prevent a loss of SHUTDOWN MARGIN or a boron dilution event so that radiological consequences from these events are precluded.

The proposed changes do not involve physical modifications to plant equipment and do not change the operational methods or procedures used for moving irradiated fuel assemblies. The proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. Relaxation of operability requirements during the specified conditions will not significantly increase the probability of occurrence of an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3–3, in which the provisions of the notes have expired, does not impact the probability or consequences of an accident previously evaluated.

Based on the above discussion, 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 involve a physical change. The change will allow the automatic start feature of systems no longer credited in the accident analyses for mitigation to be disabled through the STPNOC [STP Nuclear Operating Company] modification process. Implementation of AST provides increased operating margins for filtration system efficiencies. Application of AST provides for relaxation of certain Control Room ventilation system filtration requirements. The Fuel Handling Building filtration and holdup is no longer credited in the AST analyses. Therefore, the Fuel Handling Building Exhaust Air Ventilation system is no longer required in the Technical Specifications. It also relaxes containment integrity requirements while handling irradiated fuel and during core alterations. Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Mode 5 or Mode 6 does not create the possibility of a new or different kind of accident because these events have already been analyzed in the safety analysis with a conclusion that adequate measures exist to prevent these events.

Similarly, the proposed changes do not require any physical changes to any structures, systems or components involved in the mitigation of any accidents. Therefore, no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed changes.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3–3, in which the provisions of the notes have expired, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Approval of a change from the original source term methodology (i.e., TID 14844) to an AST methodology, consistent with the guidance in RG [NRC Regulatory Guide] 1.183, will not result in a significant reduction in the margin of safety. The safety margins and analytical conservatisms associated with the AST methodology have been evaluated and were found acceptable. The results of the revised DBA analyses, performed in support of the proposed changes, are subject to specific acceptance criteria as specified in RG 1.183. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67 and RG 1.183.

Elimination of the requirement to suspend operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or required boron concentration if the control room ventilation system is inoperable in Mode 5 or Mode 6 does not result in a reduction in a margin to safety because adequate measures exist to preclude radiological consequences from these events.

The proposed changes continue to ensure that the doses at the exclusion area boundary (EAB) and low population zone boundary (LPZ), as well as the Control Room and Technical Support Center, are within the specified regulatory limits.

Administrative changes to delete a footnote from Technical Specification surveillance requirement 4.7.7.e.3) and a note from ACTION 20 of Technical Specification Table 3.3–3, in which the provisions of the notes have expired, does not impact the margin of safety.

Therefore, based on the above discussion, 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Thomas G. Hiltz.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 21, 2007.

Description of amendment request: The license amendment request proposes revising the Technical Specification (TS) Surveillance Requirement (SR) 4.5.2.d for the inspection of Emergency Core Cooling System (ECCS) sumps for consistency with the new STP sump design. SR 4.5.2.d includes a noncomprehensive parenthetical list of sump components, some of which have been removed in the new sump screen design. The licensee proposes an administrative change to delete the parenthetical reference to sump components in its entirety.

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 is an administrative editorial change to remove unnecessary information from a surveillance requirement. It will not affect how any system, structure, or component is designed or operated and so has no potential to affect the mitigation of an accident. The change does not affect an initiator of any accident previously evaluated. Therefore, the 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 is an administrative editorial change to remove unnecessary information from a surveillance requirement. It will not affect how any system, structure, or component is designed or operated or involve any new or different plant configurations. Therefore, the change does not create the possibility of a new or different kind of accident previously evaluated.

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

Response: No.

The proposed change is editorial and administrative and consequently has no effect on the margin of safety.

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

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: June 8, 2007.

Description of amendment request: The proposed amendment would revise the technical specifications for Watts Bar Nuclear Plant, Unit 1 (WBN) to allow relaxations of various Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) logic completion times, bypass test times, allowable outage times, and surveillance testing intervals. The proposed changes implement several Technical Specifications Task Force travelers, which the NRC staff has previously reviewed and approved for incorporation into the Standard Technical Specifications for Westinghouse plants.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the *Code of Federal Regulations* (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?

The proposed changes do not result in any modifications to RTS and ESFAS hardware, design requirements, or functions. No system operational parameters are affected. The protection system will continue to perform the intended design functions consistent with the design bases and accident analyses. The proposed changes will not modify any system interfaces and, therefore, could not increase the likelihood of an accident described in the UFSAR [Updated Facility Safety Analysis Report]. The proposed amendment will not change, degrade or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR.

Plant-specific evaluations confirm the applicability of the [Westinghouse Topical Report] WCAP–14333 and WCAP–15376 analyses to WBN. Implementation of the approved changes is in accordance with the conditions of the NRC safety evaluations for these reports and will result in an insignificant risk impact.

The proposed changes to the completion time, bypass test time, and surveillance frequencies reduce the potential for inadvertent reactor trips and spurious actuations and, therefore, do not increase the probability of any accident previously evaluated. The proposed changes to the allowed completion time, bypass test time, and surveillance frequencies do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by core damage frequency [CDF] is less than 1.0E–06 per year and the impact on large early release frequency [LERF] is less than 1.0E–07 per year. In addition, for the completion time change, the incremental conditional core damage probabilities [ICCDP] and incremental conditional large early release probabilities [ICLERP] are less than 5.0E–07 and 5.0E–08, respectively. These changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by CDF, LERF, ICCDP, and ICLERP is within the acceptance criteria of existing regulatory guidance, there will not be a significant increase in the consequences of any accidents.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

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

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

The proposed amendment does not require any design changes, physical modifications or changes in normal operation of the RTS and ESFAS instrumentation. Existing setpoints will be maintained. The changes do not affect functional performance requirements of the instrumentation. No changes are required to accident analysis assumptions. The changes do not introduce different malfunctions, failure modes, or limiting single failures. The changes to the completion time, bypass test time, and surveillance frequency do not change any existing accident scenarios nor create any new or different accident scenarios.

Therefore, this 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?

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased completion time, bypass test time, and surveillance frequencies, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

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

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

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

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: June 6, 2006.

Brief description of amendment request: The proposed amendments would provide a new action for selected Technical Specifications (TSs) limiting conditions for operation to permit extension of the completion times of action requirements, provided risk is assessed and managed. A new program, the Configuration Risk Management Program, would be added to the Administrative Controls of TSs.

Date of publication of individual notice in **Federal Register**: June 12, 2007.

Expiration date of individual notice: July 12, 2007.

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 51.22(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 (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 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 PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: November 13, 2006.

Brief description of amendment: The proposed amendment revises the technical specification (TS) testing frequency for the surveillance requirement (SR) in TS 3.2.4, ''Control Rod Scram Times.'' Specifically, the proposed change would revise the frequency for SR 3.1.4.2, control rod scram time testing, from ''120 days cumulative operation in MODE 1,'' to ''200 days cumulative operation in MODE 1.'' This operating license improvement was made available by the Nuclear Regulatory Commission on August 23, 2004, as part of the consolidated line item improvement process.

Date of issuance: July 5, 2007.

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

Amendment No.: 177.

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

Date of initial notice in ***Federal Register:*** April 10, 2007 (72 FR 17944). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 5, 2007.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: June 14, 2006, as supplemented by letters dated November 27, 2006 and January 17, 2007.

Brief description of amendment: The amendment revised the technical specifications (TSs) to allow a one-time change in the Appendix J, Type A, Containment Integrated Leak Rate Test from the required 10 years to 15 years.

Date of issuance: June 29, 2007.

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

Amendment No. 239.

Facility Operating License No. NPF–49: Amendment revised the technical specifications.

Date of initial notice in **Federal Register:** September 12, 2006 (71 FR 53717). The November 27, 2006 and January 17, 2007, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendment: June 2, 2006.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to incorporate revised requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 20. Specifically, the amendment revises the definitions for Members of the Public and Unrestricted Area, adds a definition for Restricted Area, revises the requirements for limitations on the concentrations of radioactive material released in liquid and gaseous effluents, and revises the references for radioactive effluent control requirements.

Date of issuance: June 29, 2007.

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

Amendment Nos.: 187 and 148.

Facility Operating License Nos. NPF–39 and NPF–85: This amendment revised the license and Technical Specifications.

*Date of initial notice in **Federal Register:*** April 10, 2007 (72 FR 17949). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: June 8, 2006, as supplemented by letter dated February 5, 2007.

Brief description of amendments: These amendments modify the Technical Specifications by removing reference to ''the Banked Position Withdrawal Sequence'' and replace it with ''the analyzed rod position sequence.''

Date of issuance: June 29, 2007.

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

Amendments Nos.: 260 and 264.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the License and Technical Specifications.

*Date of initial notice in **Federal Register:*** August 15, 2006 (71 FR 46934). The February 5, 2007, letter, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2007.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: July 14, 2006, as supplemented by letter dated June 5, 2007.

Brief description of amendments: The proposed changes modified Technical Specification (TS) requirements related to required end states for TS action statements that are consistent with the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Change Traveler, TSTF–423, ''Risk Informed Modification to Selected Required Action End States for BWR [boiling-water reactor] Plants.''

Date of issuance: July 12, 2007.

Effective date: As of the date of issuance, to be implemented within 120 days.

Amendments Nos.: 261 and 265.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the TSs.

*Date of initial notice in **Federal Register:*** December 19, 2006 (71 FR 75994). The letter dated June 5, 2007, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 19, 2007.

Brief description of amendment: The amendment modifies the technical specifications requirements for the diesel fuel oil program by relocating references to specific standards for fuel oil testing to licensee-controlled documents and adds alternate criteria to the ''clear and bright'' acceptance test for new fuel oil.

Date of issuance: July 12, 2007.

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

Amendment No.: 146.

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

*Date of initial notice in **Federal Register:*** April 10, 2007 (72 FR 17950). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: January 30, 2007, supplemented by your letter dated April 11, 2007.

Brief description of amendment request: The amendments revise Section 5 of the technical specifications to reflect the move to a site vice president organizational structure for Joseph M. Farley Nuclear Plant, Units 1 and 2.

Date of issuance: July 16, 2007.

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

Amendment Nos.: 175, 168.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the technical specifications.

*Date of initial notice in **Federal Register:*** February 13, 2007 (72 FR 6790). The supplement provided clarifying information that did not change the scope of the 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 July 16, 2007.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: August 2, 2004, as resubmitted on June 6, 2006, and supplemented by letters dated December 28, 2006, February 28, May 9, and May 17, 2007.

Brief description of amendments: The amendments provide for a new action for selected Technical Specifications (TS) limiting conditions for operation to permit extending the completion times allowed for action requirements subject to the requirements that the risk is assessed and managed. A new Configuration Risk Management Program is added to the TS under Administrative Controls, as a risk assessment tool.

Date of issuance: July 13, 2007.

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

Amendment Nos.: Unit 1—179; Unit 2—166.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Facility Operating Licenses and Technical Specifications.

*Date of initial notice in **Federal Register:*** June 12, 2007 (72 FR 32332). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 13, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 19th day of July 2007.

For the Nuclear Regulatory Commission

Catherine Haney,

Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E7–14350 Filed 7–30–07; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide: Issuance, Availability

AGENCY: Nuclear Regulatory Commission.

ACTION: Draft Regulatory Guide: Issuance, Availability.

FOR FURTHER INFORMATION CONTACT: NRC Senior Program Manager, Satish Aggarwal, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: (301) 415–6005 or e-mail *SKA@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

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

The draft regulatory guide, entitled ''Qualification of Safety-Related Battery Chargers & Inverters for Nuclear Power Plants,'' is temporarily identified by its task number, DG–1148, which should be mentioned in all related correspondence.

The Commission's regulations in Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR part 50), ''Domestic Licensing of Production and Utilization Facilities,'' require that structures, systems, and components that are important to safety in a nuclear power plant must be designed to accommodate the effects of environmental conditions [i.e., remain functional under postulated design-basis events (DBEs)]. Toward that end, the general requirements are contained in General Design Criteria 1, 2, 4, and 23 of Appendix A, ''General Design Criteria for Nuclear Power Plants,'' to 10 CFR part 50. Augmenting those general requirements, the specific requirements pertaining to qualification of certain electrical equipment important to safety are contained in 10 CFR 50.49, ''Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.'' In addition, Criterion III, ''Design Control,'' of Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants,'' to 10 CFR part 50, requires that where a test program is used to verify the adequacy of a specific design feature, it should include suitable qualification testing of a prototype unit under the most severe DBE.

This regulatory guide describes a method that the NRC considers acceptable for use in implementing specific parts of the agency's regulations for qualification of safety-related battery chargers and inverters for nuclear power plants.

II. Further Information

The NRC is soliciting comments on Draft Regulatory Guide DG–1148. Comments may be accompanied by relevant information or supporting data, and should mention DG–1148 in the subject line. Comments submitted in writing or in electronic form will be made available to the public in their entirety through the NRC's Agencywide Documents Access and Management System (ADAMS). Personal information will not be removed from your comments. You may submit comments by any of the following methods:

1. *Mail comments to:* Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

2. *E-mail comments to: NRCREP@nrc.gov.* You may also submit comments via the NRC's rulemaking Web site at *http://ruleforum.llnl.gov.* Address questions about our rulemaking Web site to Carol A. Gallagher (301) 415–5905; e-mail *CAG@nrc.gov.*

3. *Hand-deliver comments to:* Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

4. *Fax comments to:* Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission at (301) 415–5144.

Requests for technical information about Draft Regulatory Guide DG–1148 may be directed to NRC Senior Program Manager, Satish Aggarwal, at (301) 415–6005 or e-mail *SKA@nrc.gov.*

Comments would be most helpful if received by October 2, 2007. Comments received after that date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date. Although a time limit is given, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Electronic copies of Draft Regulatory Guide DG–1148 are available through the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/doc-collections/.* Electronic copies are also available in ADAMS (*http://www.nrc.gov/reading-rm/adams.html*), under Accession No. ML071440292.

In addition, regulatory guides are available for inspection at the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland. The PDR's mailing address is USNRC PDR, Washington, DC 20555–0001. The PDR can also be reached by telephone at (301) 415–4737 or (800) 397–4205, by fax at (301) 415–3548, and by e-mail to *PDR@nrc.gov.*

Please note that the NRC does not intend to distribute printed copies of Draft Regulatory Guide DG–1148, unless specifically requested on an individual basis with adequate justification. Such requests for single copies of draft or final guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions 3 should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Reproduction and Distribution Services Section; by e-mail to *DISTRIBUTION@nrc.gov;* or by fax to (301) 415–2289. Telephone requests cannot be accommodated.

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

Dated at Rockville, Maryland, this 25 day of July, 2007.

For The Nuclear Regulatory Commission.

Andrea Valentin,

Chief, Regulatory Guide Branch, Division of Fuel, Engineering and Radiological Research, Office of Nuclear Regulatory Research.

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