performance of the functions of the NSF, including whether the information shall have practical utility; (b) the accuracy of the NSF's estimate of the burden of the proposed collection of information; (c) ways to enhance the quality, utility, and clarity of the information on respondents, including through the use of automated collection techniques or other forms of information technology; (d) ways to minimize the burden of the collection of information on those who are to respond, including through the use of appropriate automated, electronic, mechanical or other technological collection techniques or other forms of information technology.

Dated: August 4, 2010.

Suzanne H. Plimpton,

Reports Clearance Officer, National Science Foundation.

[FR Doc. 2010–19626 Filed 8–9–10; 8:45 am]

BILLING CODE 7555–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2010–0272]

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) 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 15, 2010 to July 28, 2010. The last biweekly notice was published on July 27, 2010 (75 FR 44020).

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 Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

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

Normally, the Commission will not issue the amendment until the expiration of 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, Rules, Announcements and Directives Branch (RADB), TWB–05–B01M, 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 faxed to the RADB at 301–492–3446. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. 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 person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Room O1F–21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or 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 identify the specific contentions which the requestor/petitioner 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 requestor/petitioner 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 requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner 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 requestor/petitioner to relief. A requestor/petitioner 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, 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.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least ten (10) days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at *hearing.docket@nrc.gov,* or by telephone at (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.* System requirements for accessing the E–Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through EIE, users will be required to install a Web browser plug-in from the NRC Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html.*

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html.* A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html,* by e-mail at *MSHD.Resource@nrc.gov,* or by a toll-free call at (866) 672–7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted 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; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at *http://ehd.nrc.gov/EHD_Proceeding/home.asp,* unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Room O1–F21, 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.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr.resource@nrc.gov.*

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendment requests: April 29, 2010, as supplemented by letter dated July 22, 2010.

Description of amendment requests: The proposed change will add to Technical Specification 5.6.5.b an additional topical report describing an NRC reviewed and approved analytical method for determining core operating limits. The new analytical method, which is described in AREVA Topical Report ANP–10298PA, ACE/ATRIUM 10XM Critical Power Correlation, Revision 0, March 2010, provides a new correlation for predicting the critical power for boiling water reactors containing ATRIUM 10XM fuel.

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 proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. The proposed amendments support the use of the AREVA ATRIUM 10XM fuel design at BSEP [Brunswick Steam Electric Plant]. The addition of an approved analytical methodology in Technical Specification Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The NRC-approved methodology ensures that the output accurately models core behavior. Since no individual precursors of an accident are affected, the proposed amendments do not increase the probability of a previously analyzed event.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods used to establish core operating limits. The addition of the topical report to Technical Specification 5.6.5.b will allow a new analytical methodology to be used to determine critical power ratio limits. Minimum Critical Power Ratio (MCPR) Safety Limit values, which are defined in Technical Specification 2.1.1.2, are calculated to ensure that greater than 99.9 percent of the fuel rods in the reactor core avoid transition boiling during plant operation, if the safety limit is not exceeded. The derivation of MCPR Safety Limit values in the Technical Specifications, using these NRC-accepted methods, will continue to ensure the MCPR Safety Limit is not exceeded during all modes of plant operation and anticipated operational occurrences. The addition of the analytical methodology described in Topical Report ANP–10298PA to Technical Specification 5.6.5.b does not alter the assumptions of accident analyses or the Technical Specification Bases. Based on the above, the proposed amendments do not increase the consequences of a previously analyzed accident.

Therefore, the proposed amendments 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.

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 amendments do not involve any plant configuration modifications, do not involve any changes to allowable modes of operation, and do not introduce any new failure mechanisms. The proposed topical report addition to Technical Specification 5.6.5.b provides an analytical methodology for determining core critical power limits that ensures no new accident precursors are created.

Therefore, the proposed amendments 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 proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. This addition to Technical Specification 5.6.5.b will allow a new NRC-accepted analytical methodology to be used to determine critical power ratio limits. The MCPR Safety Limit provides a margin of safety by ensuring that at least 99.9 percent of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences if the MCPR Safety Limit is not exceeded. The proposed change will ensure the current level of fuel protection is maintained by continuing to ensure that the fuel design safety criterion (*i.e.,* that no more than 0.1 percent of the rods are expected to be in boiling transition if the MCPR Safety Limit is not exceeded) is met.

Therefore, the proposed amendments do not result in 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, NC 27602.

NRC Branch Chief: Douglas A. Broaddus.

Carolina Power and Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendment requests: April 29, 2010, as supplemented by letter dated July 22, 2010.

Description of amendment requests: The proposed change would add, to Technical Specification 5.6.5.b, an additional topical report describing an NRC reviewed and approved analytical method for determining core operating limits. The new analytical method, which is described in AREVA Topical Report BAW–10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008, provides a new statistical thermal-mechanical evaluation methodology for determining reactor core linear heat generation limits in boiling water reactors.

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 proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. The proposed amendments support the use the AREVA ATRIUM 10XM fuel design at BSEP [Brunswick Steam Electric Plant]. The addition of an approved analytical methodology in Technical Specification Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The NRC-approved methodology ensures that the output accurately models core behavior. Since no individual precursors of an accident are affected, the proposed amendments do not increase the probability of a previously analyzed event.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods used to establish core operating limits. The addition of the topical report to Technical Specification 5.6.5.b will allow a new thermal-mechanical methodology, based on the RODEX4 fuel performance code, to be used to determine reactor core linear heat generation rate limits monitored as specified by Technical Specification 3.2.3. The addition of the analytical methodology described in Topical Report BAW–10247PA to Technical Specification 5.6.5.b does not alter the assumptions of accident analyses or the Technical Specification Bases. Based on the above, the proposed amendments do not increase the consequences of a previously analyzed accident.

Therefore, the proposed amendments 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.

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 amendments do not involve any plant configuration modifications, do not involve any changes to allowable modes of operation, and do not introduce any new failure mechanisms. The proposed topical report addition to Technical Specification 5.6.5.b provides an analytical methodology for determining reactor core linear heat generation rate limits that ensures no new accident precursors are created.

Therefore, the proposed amendments 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 proposed amendments add an additional analytical methodology to the list of NRC-approved analytical methods identified in Technical Specification 5.6.5.b that can be used to establish core operating limits. This addition to Technical Specification 5.6.5.b will allow a new NRC-accepted analytical methodology to be used to determine reactor core linear heat generation rate limits.

Limits on the linear heat generation rate are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences. Exceeding the linear heat generation rate limit could potentially result in fuel damage and subsequent release of radioactive materials. The mechanisms that could cause fuel damage during normal operations and operational transients and that are considered in fuel evaluations are rupture of the fuel rod cladding caused by strain and overheating of the fuel. The proposed change will ensure the current level of fuel protection is maintained (*i.e.,* that the fuel design safety criteria of less than one percent plastic strain of the fuel cladding is met and incipient centerline melting of the fuel does not occur) and thus assure that rupture of the fuel rod cladding caused by strain and overheating of the fuel does not occur.

Therefore, the proposed amendments do not result in 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, NC 27602.

NRC Branch Chief: Douglas A. Broaddus.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: June 10, 2009, supplemented by letters dated September 16, 2009, and July 23, 2010.

Description of amendment request: The proposed amendment would revise Fermi 2 Plant Operating License, Appendix A, Technical Specification (TS) Table 3.3.8.1–1, Function 2 (Degraded Voltage) to identify an additional time delay logic for Loss-of-Coolant Accident (LOCA) concurrent with degraded voltage conditions. Specifically, this proposed amendment adds a new time delay logic associated with Function 2 for a degraded voltage concurrent with a LOCA. This will bring Fermi 2 into compliance with 10 CFR Part 50, Appendix A, General Design Criterion (GDC)—17, "Electric Power Systems." In addition, it would revise the TS maximum and minimum allowable values for the 4.16kV Emergency Bus Undervoltage (Degraded Voltage) and revise the minimum Emergency Diesel Generator (EDG) output voltage acceptance criterion in Surveillance Requirements (SRs) 3.8.1.2, 3.8.1.7, 3.8.1.10, 3.8.1.11, 3.8.1.14, and 3.8.1.17. The additional changes resulted from a reconstitution effort of the electrical design bases calculations to support the backfit modifications, necessary to address issues identified in the Component Design Bases Inspection (CDBI) at Fermi 2. This notice supersedes the notice published in the **Federal Register** on August 11, 2009, (74 FR 40235), 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Providing the additional logic ensures the timely transfer of plant safety system loads to the Emergency Diesel Generators in the event a sustained degraded bus voltage is present with a Loss of Coolant Accident (LOCA) signal. This ensures that under these degraded bus voltage conditions, Emergency Core Cooling System (ECCS) equipment is powered from the emergency diesel generators in a timely manner. This change is needed to bring Fermi 2 into full compliance with 10 CFR Part 50, Appendix A, General Design Criterion-17, "Electric Power Systems," and to meet the requirements of NUREG–0800 Rev. 2, Branch Technical Position (BTP) Power Systems Branch (PSB)-1. The time delay supports the time assumed in the accident analysis for water injection into the reactor vessel under LOCA conditions.

The proposed TS change to the maximum and minimum allowable voltages for the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) affects the separation of an Emergency Bus that is experiencing degraded voltage from the offsite power system and the transfer to an emergency diesel generator. While the allowed voltage range is narrower, the function remains the same. The narrower voltage range has been analyzed and is needed to ensure spurious trips are avoided. The proposed change does not affect any accident initiators or precursors. As a result, the probability of any accident previously evaluated is not significantly increased.

The consequences of any accident previously evaluated are not increased since the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) relays will continue to meet their required function to transfer the 4160 volt Emergency Buses to the emergency diesel generators in the event of a degraded voltage condition on the offsite power supply. This transfer ensures that the electrical equipment is capable of performing its intended function to meet the requirements of the accident analyses.

The increase in the minimum EDG output voltage acceptance criterion value in TS 3.8.1 surveillance requirements does not adversely affect any of the parameters in the accident analyses. The change increases the minimum allowed EDG output voltage acceptance criterion to ensure that sufficient voltage is available to operate the required Emergency Safety Feature (ESF) equipment under accident conditions. The increase in the minimum allowed EDG output voltage in the TS surveillance requirements ensures that adequate voltage is available to support the assumptions made in the Design Bases Accident (DBA) analyses. DBA analyses assume that onsite standby emergency power will provide an adequate power source to operate safe shutdown equipment and to mitigate consequences of design bases accidents. This conservative change of the acceptance criterion enhances the testing requirements of the onsite emergency diesel generators and ensures the reliability of this power source. Changing the acceptance criterion does not affect the probability of evaluated accidents and it provides better assurance of EDG reliability in mitigating consequences of accidents.

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

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

The proposed change does not affect any of the current degraded voltage logic schemes or any other equipment provided to mitigate accidents. It utilizes existing logic systems to isolate safety buses from the grid and re-power those safety buses using the onsite emergency power system. The change utilizes a narrower voltage range and a shorter time delay to ensure that in the case of a sustained degraded voltage condition concurrent with a LOCA signal, the safety electrical power buses will be transferred from the offsite power system to the onsite power system in a timely manner to ensure water is injected into the reactor vessel in the time assumed and evaluated in the accident analysis.

No new or different accidents result from the proposed change. The proposed TS change to the maximum and minimum allowable voltages for the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) does not affect existing accident precursors or modes of operation nor does it introduce new ones. The relays will continue to detect degraded voltage conditions and transfer the Emergency Buses to their respective emergency diesel generators in time to ensure adequate voltage is available for proper safety equipment performance, and to prevent equipment damage. The function of the relays remains the same.

The change in the value of the minimum EDG output voltage acceptance criterion supports the assumptions in the accident analyses that sufficient voltage will be available to operate ESF equipment on the Class 1E buses when these buses are powered from the onsite emergency diesel generators. The maximum EDG output voltage of 4580 volts is not affected by this change. The change in the minimum EDG output voltage from 3873 to 3950 volts ensures the reliability of the onsite emergency power source.

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

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

The proposed change implements a new design for a reduced time delay to isolate safety buses from offsite power if a Loss of Coolant Accident were to occur concurrent with a sustained degraded voltage condition and uses a narrower voltage range for degraded bus undervoltage. This ensures that emergency core cooling system pumps inject water into the reactor vessel within the time assumed and evaluated in the accident analysis, consistent with the requirements of BTP PSB–1 Section B.1.b. and 10 CFR Part 50, Appendix A, General Design Criterion-17, "Electric Power Systems."

The proposed TS change to the maximum and minimum allowable voltages for the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) will allow all safety loads to have sufficient voltage to perform their intended safety functions while ensuring spurious trips are avoided. Thus, the results of the accident analyses will not be affected as the input assumptions are protected.

The proposed TS change for the maximum allowable values for the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) provides a greater margin between the predicted worst case transient voltages and the maximum reset value of the degraded voltage relays. This change increases the probability that the offsite power source remains available and connected to the auxiliary power system during postulated transients. The analytical limit voltage for the safety related 4160 volt buses is unchanged and the proposed TS changes for the minimum allowable values for the 4160 volt Emergency Bus Undervoltage (Degraded Voltage) still ensures that this limit is protected. This is consistent with the requirements of 10 CFR Part 50, Appendix A, General Design Criterion-17, "Electric Power Systems."

The proposed change in the minimum EDG output voltage acceptance criterion in TS 3.8.1 surveillance requirements does not affect the surveillance frequency or different testing requirements, only the acceptance criterion. The change provides a better assurance that the onsite power source is able to satisfy the design requirements assumed in the accident analyses to mitigate the consequences of design bases accidents.

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: David G. Pettinari, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Branch Chief: Robert J. Pascarelli.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: June 1, 2010.

Description of amendment request: The licensee proposed to revise the Kewaunee licensing basis, approving the licensee to operate the load tap changers (LTCs) on two new transformers to operate in the automatic mode. The LTCs are subcomponents of the two new transformers, one has already been installed and one to be installed. The LTCs are designed to compensate for potential offsite power voltage variations and will provide added assurance that acceptable voltage is maintained for safety-related equipment.

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 (NSHC). The NRC staff reviewed the licensee's NSHC analysis and has prepared its own as follows:

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

Response: No.

The function of the LTCs is to ensure that acceptable voltage is maintained for safety-related equipment. The only postulated accident previously evaluated where the probability of occurrence may be potentially affected by operating the LTCs in automatic mode is the loss of offsite power (LOOP) accident. However, the licensee's analysis shows that, as a result of availability of backup equipment and systems, the probability of a LOOP would not be increased by operation of the LTCs in the automatic mode. Furthermore, operation of the LTCs in the automatic mode is not likely to degrade the Kewaunee electrical system; thus, the electrical system will continue to fulfill its design functions during normal and accident conditions. As a result, operating the LTCs in automatic mode will not be a factor to increase the consequences of previously evaluated accidents. In summary, the probability of occurrence and the consequences of the previously analyzed accidents would not be affected in any way by the proposed licensing basis change.

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

Response: No.

Other than the installation of the two new transformers (which is not the subject of the proposed amendment), the proposed change of licensing basis to allow the LTCs to be operated in the automatic mode does not involve any physical alteration of the plant, nor does it change methods and procedures governing plant operation. The proposed change will not impose any new or eliminate any old safety requirements on the plant electrical system.

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 has no effect on any safety analysis methods, scenarios, or assumptions involving the electrical system.

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

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 proposed amendment involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., Counsel for Dominion Energy Kewaunee, Inc., 120 Tredegar Street, Richmond, VA 23219.

NRC Branch Chief: Robert J. Pascarelli.

Duke Energy Carolinas, LLC, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2 (Catawba), York County, South Carolina; Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2 (McGuire), Mecklenburg County, North Carolina

Date of amendment request: December 14, 2009.

Description of amendment request: The amendments would revise the Technical Specifications Section 3.8.4 "DC [Direct Current] Sources—Operating" Surveillance Requirements 3.8.4.2 and 3.8.4.5 for McGuire and 3.8.4.3 and 3.8.4.6 for Catawba.

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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Performing the battery Surveillances is not an initiator to any accident sequence previously evaluated in the Updated Final Safety Analysis Report. The Batteries are still required to be operable, meet the Surveillance Requirements, and be capable of performing any mitigation function as designed. Revising the battery Surveillance resistance values and adding the total average resistance limit, as supported by calculations, will help ensure that the voltage and capacity of the Batteries remain within the design basis.

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

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

Response: No.

This amendment does not involve a modification to the plant or a change in how the plant is operated. No new accident causal mechanisms are created as a result of this proposed amendment. No changes are being made to any structure, system, or component which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators and does not impact any safety analysis.

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 amendment involve a significant reduction in the margin of safety?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fuel cladding, reactor coolant and containment systems will not be impacted by the proposed change. The proposed McGuire and Catawba battery connection resistance limits ensure the continued availability and operability of the Batteries. As such, sufficient DC capacity to support operation of mitigation equipment remains within the design basis.

Therefore, it is concluded that 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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Gloria Kulesa.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 23, 2010.

Description of amendment request: The current Arkansas Nuclear One, Unit No. 2 Technical Specification (TS) 6.5.8, "Inservice Testing Program," contains references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI as the source of requirements for the inservice testing (IST) of ASME Code Class 1, 2, and 3 pumps and valves. The proposed amendment would delete the references to Section XI of the ASME Code and incorporate references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code). The proposed amendment would also correct some nonstandard frequencies utilized in the IST Program in which the provisions of Surveillance Requirement 3.0.2 are applicable. The proposed changes are consistent with Technical Specification Task Force (TSTF) Technical Change Travelers 479–A, "Changes to Reflect Revision to 10 CFR 50.55a," and 497–A, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less."

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 revises TS 6.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code which are consistent with the expectations of 10 CFR 50.55a.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed change does not involve the addition or removal of any equipment, or any design changes to the facility.

Therefore, this proposed change does not represent 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 does not involve a modification to the physical configuration of the plant (*i.e.,* no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change does not introduce a new accident initiator, accident precursor, or malfunction mechanism.

Therefore, this proposed change does not create the possibility of an accident or a different kind than previously evaluated.

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

Response: No.

The proposed change revises TS 6.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code, which are consistent with the expectations of 10 CFR 50.55a. The safety function of the affected pumps and valves are maintained.

Therefore, this 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: Joseph A. Aluise, Associate General Counsel—Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

Luminant Generation Company LLC, Docket Nos. 50–445 and 50–446, Comanche Peak Nuclear Power Plant, Units 1 and 2, Somervell County, Texas

Date of amendment request: May 27, 2010.

Brief description of amendments: The proposed amendments would revise the Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2, Technical Specification (TS) 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," by relocating the current stored diesel fuel oil and lube oil numerical volume requirements from the TS to the TS Bases so that it may be modified under licensee control. The TS would be modified so that the stored diesel fuel oil and lube oil inventory will require that a 7-day supply be available for each diesel generator. Condition A and Condition B in the Action table and Surveillance Requirements (SRs) 3.8.3.1 and 3.8.3.2 would also be revised to reflect the above change. The proposed changes are consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Revision 1 to Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler 501, "Relocate Stored Fuel Oil and Lube Oil Volume Values to Licensee Control."

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 relocates the volume of diesel fuel oil and lube oil required to support 7-day operation of the onsite diesel generators, and the volume equivalent to a 6-day supply for fuel oil and, for lube oil, a 2-day supply to licensee control. The specific volume of fuel oil equivalent to a 7- and 6-day supply is calculated using the NRC-approved methodology described in Regulatory Guide 1.137, Revision 1, "Fuel-Oil Systems for Standby Diesel Generators" and ANSI [American National Standards Institute] N195 1976, "Fuel Oil Systems for Standby Diesel-Generators." The CPNPP specific volumetric requirements for lube oil were originally based on the manufacturer's consumption values; however, the volumetric requirements have been refined over time based on actual plant data and engine performance. As approved in CPNPP TS License Amendment 75, the current lube oil volumetric requirements are based on the diesel generator lube oil consumption rate, avoidance of vortexing, static versus run lube oil level changes, and volume versus tank level data.

Therefore, this proposed change is consistent with TSTF–501 as approved by the NRC. Because the requirement to maintain a 7-day supply of diesel fuel oil and lube oil is not changed and is consistent with the assumptions in the accident analyses, and the actions taken when the volume of fuel oil and lube oil are less than a 6-day and 2-day supply have not changed, neither the probability or the consequences of any accident previously evaluated will be affected.

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 change does not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis but ensures that the diesel generator operates as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

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

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

Response: No.

The proposed change relocates the volume of diesel fuel oil and lube oil required to support 7-day operation of the onsite diesel generators, and the volume equivalent to a 6- and 2- (for fuel oil and lube oil, respectively) day supply to licensee control. As the bases for the existing limits on diesel fuel oil and lube oil are not changed, no change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change.

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: Timothy P. Matthews, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: November 23, 2009, as supplemented on December 11 and December 18, 2009, and July 23, 2010 (TS 09–06).

Description of amendment request: On March 27, 2009, the **Federal Register** Notice 74 FR 13926 issued the final rule that amended Title 10 of the *Code of Federal Regulations* (10 CFR), Part 73, "Physical Protection of Plants and Materials." Specifically, the regulations in 10 CFR 73.54 "Protection of Digital Computer and Communication Systems and Networks" establish the requirements for a cyber security program to protect digital computer and communication systems and networks against cyber attacks. The proposed amendment would include the proposed Cyber Security Plan, its implementation schedule, and a revised Physical Protection license condition for Sequoyah Nuclear Plant, Units 1 and 2 to fully implement and maintain in effect all provisions of the Nuclear Regulatory Commission approved Cyber Security Plan as required by 10 CFR 73.54.

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:

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

Neither the proposed additional license condition nor the Cyber Security Plan directly impacts the physical configuration or function of plant structures, systems, or components (SSCs). Likewise, they do not change the manner in which SSCs are operated, maintained, modified, tested, or inspected. Neither the proposed additional license condition nor the Cyber Security Plan introduces any initiator of any accident previously evaluated. Any modifications to the physical configuration or function of SSCs or the manner in which SSCs are operated, maintained, modified, tested, or inspected that might result from the implementation of the Cyber Security Plan will be fully evaluated by existing regulatory processes (*e.g.,* 10 CFR 50.59) prior to their implementation to ensure that they do not result in the probability or consequences of an accident previously evaluated.

Therefore, it is concluded that this amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed amendment is intended to provide high assurance that safety-related SSCs are protected from cyber attacks. Inclusion of the additional condition in the Facility Operating License to implement the Cyber Security Plan does not directly alter the plant configuration, require new plant equipment to be installed, alter or create new accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected.

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

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

The proposed amendment does not involve any physical changes to plant or alter the manner in which plant systems are operated, maintained, modified, tested, or inspected. The proposed change does 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 affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. Adding a license condition to require implementation of Cyber Security Plan will not reduce a margin of safety because the requirements of the Plan are designed to provide high assurance that safety-related SSCs are protected from cyber attacks.

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

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

NRC Branch Chief: Douglas A. Broaddus.

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

Date of amendment request: March 30, 2010.

Description of amendment request: This amendment request involves the adoption of approved changes to the Standard Technical Specifications (STSs) for Westinghouse Pressurized Water Reactors (NUREG–1431), to allow relocation of specific TS surveillance frequencies to a licensee-controlled program. The proposed changes are described in Technical Specification Task Force (TSTF) Traveler, TSTF–425, Revision 3 (ADAMS Accession No. ML090850642) related to the "Relocation of Surveillance Frequencies to Licensee Control—Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b," and are described in the Notice of Availability published in the **Federal Register** on July 6, 2009 (74 FR 31996). The proposed changes are consistent with NRC-approved Industry/TSTF Traveler, TSTF–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-[RITSTF] Initiative 5b." The proposed changes relocate surveillance frequencies to a licensee-controlled program, the Surveillance Frequency Control Program (SFCP). The changes are applicable to licensees using probabilistic risk guidelines contained in NRC-approved NEI 04–10, "Risk-Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. 071360456). In addition, administrative/editorial deviations of the TSTF–425 inserts and the existing TS wording are being proposed to fit the custom TS format.

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 any accident previously evaluated?

Response: No.

The proposed changes relocate the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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.

No new or different accidents result from utilizing the proposed changes. The changes do not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Dominion will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04–10, Rev. 1 in accordance with the TS SFCP. NEI 04–10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Resources Services, Inc., 120 Tredegar St., RS–2, Richmond, VA 23219.

NRC Branch Chief: Gloria Kulesa.

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, Room O1–F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System (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.resource@nrc.gov.*

Carolina Power and Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: October 27, 2009.

Brief Description of amendments: The proposed amendments modified technical specifications (TSs) requirements related to primary containment isolation instrumentation in accordance with the Nuclear Regulatory Commission-approved Technical Specification Task Force (TSTF), Standard Technical Specifications Change Traveler, TSTF–306, Revision 2, "Add action to LCO [Limiting Condition for Operation] 3.3.6.1 to give option to isolate the penetration." The proposed amendment would revise TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation," by adding an ACTIONS note allowing intermittent opening, under administrative control, of penetration flow paths that are isolated. Additionally, the traversing in-core probe system would be added as a separate isolation function with an associated Required Action to isolate the penetration within 24 hours rather than immediately initiating a unit shutdown.

Date of issuance: July 23, 2010.

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

Amendment Nos.: 255 and 283.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments changed the Technical Specifications.

Date of initial notice in ***Federal Register****:* January 26, 2010 (75 FR 4114).

The Commission's related evaluation of the amendments is contained in the Safety Evaluation dated July 23, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 27, 2010, as supplemented by letter dated March 22, 2010.

Brief description of amendment: The amendment revises a Limiting Condition for Operation (LCO) in Technical Specifications (TS) Section 3.6.2.2.a to incorporate an expanded range of eductor flow rates for the Containment Spray Additive System as a result of the use of a new chemical model and new boric acid equilibrium data, revised sump pH limits, and changes to the Containment Spray Additive Tank concentration and volume limits.

Date of issuance: July 16, 2010.

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

Amendment No. 134.

Renewed Facility Operating License No. NPF–63: The amendment revises the technical specifications and facility operating license.

Date of initial notice in ***Federal Register***: March 23, 2010 (75 FR 13788). The supplement dated March 22, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on March 23, 2010 (75 FR 13788).

The Commission's related evaluation of the amendment is contained in a safety evaluation dated July 16, 2010.

No significant hazards consideration comments received: No.

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

Date of application of amendments: August 6, 2009, as supplemented by letter dated February 23, 2010.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) by changing the surveillance requirement frequency for TS 3.4.12, "Low Temperature Overpressure Protection System," from 6 months to 18 months.

Date of Issuance: July 21, 2010.

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

Amendment Nos.: 368, 370, and 369.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the licenses and the TSs.

Date of initial notice in ***Federal Register***: March 9, 2010 (75 FR 10827). The supplement dated February 23, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 2010.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: November 23, 2009, as superseded on March 18, 2010, as supplemented on May 11 and June 3, 2010.

Brief description of amendment: The amendment revises TS Surveillance Requirements (SRs) 3.4.3.2 and 3.5.1.13 by deleting the current requirement to manually actuate each main steam safety/relief valve (SRV) during plant startup. SRs 3.4.3.2 and 3.5.1.13 have been modified to require that the SRVs be tested in accordance with the inservice test program that meets the requirements of American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plants.

Date of issuance: July 21, 2010.

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

Amendment No.: 297.

Renewed Facility Operating License No. DPR–59: The amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 20, 2010 (75 FR 20631). The May 11 and June 3, 2010, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 2010.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant (JAFNPP), Oswego County, New York

Date of application for amendment: July 31, 2009, as supplemented by letters dated March 5 and June 17, 2010.

Brief description of amendment: The change revised the JAFNPP Technical Specifications (TSs) Surveillance Requirements (SRs) for testing of the Residual Heat Removal System Shutdown Cooling (SDC) mode Containment Isolation, Reactor Pressure—High Function by replacing the current requirement to perform TS SR 3.3.6.1.3, Perform Channel Calibration, with TS SR 3.3.6.1.1 Perform Channel Check, SR 3.3.6.1.2, Perform Channel Functional Test, SR 3.3.6.1.4, Calibrate the Trip Units, and SR 3.3.6.1.5, Perform Channel Calibration. These changes are to support a proposed plant modification to increase the reliability of SDC isolation logic by changing the source of the reactor high pressure input signal.

Date of issuance: July 21, 2010.

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

Amendment No.: 298.

Renewed Facility Operating License No. DPR–59: The amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register***: October 6, 2009 (74 FR 51239).

The supplements dated March 5 and June 17, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 28, 2009, as supplemented on February 25, 2010, and May 24, 2010.

Brief description of amendments: The amendment modifies the PBAPS Unit 2 Technical Specification (TS) Section 5.5.12 to reflect a one-time extension of the Type A containment Integrated Leak Rate Test (ILRT) to no later than October 2015. The TS revision allows a one-time extension of 5 years to the 10-year frequency of the performance-based leakage rate testing program for the PBAPS Unit 2 containment Type A ILRT test.

Date of issuance: July 20, 2010.

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

Amendment Nos.: 276.

Renewed Facility Operating License Nos. DPR–44: Amendment revised the Technical Specifications.

Date of Initial Notice in ***Federal Register***: May 18, 2010 (75 FR 27830).

The supplements dated February 25, 2010, and May 24, 2010, clarified the application, did not expand the scope of the application as originally noticed, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 20, 2010.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), Oswego County, New York

Date of application for amendment: September 18, 2009, as supplemented on October 15, 2009, and April 14, 2010.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) by modifying TS

Section 3.2.7.1 and 4.2.7.1, "Primary Coolant System Pressure Isolation Valves," to incorporate requirements that are consistent with Section 3.4.5 of the Improved Standard Technical Specifications, NUREG–1433, Revision 3.0, "Standard Technical Specifications General Electric Plants, BWR/4."

Date of issuance: July 26, 2010.

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

Amendment No.: 206.

Renewed Facility Operating License No. DPR–63: The amendment revises the License and TSs.

Date of initial notice in ***Federal Register:*** October 14, 2009 (74 FR 52824). The supplemental letters dated October 15, 2009, and April 14, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination noticed in the **Federal Register** on October 14, 2009 (74 FR 52824).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 2010.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of application for amendment: April 9, 2010, and supplemented May 7, 2010.

Brief description of amendment: The amendment Request deletes Technical Specification 3.1.3, "Fuel Storage Pool Liner Water Level." Additional conforming and administrative changes are also made.

Date of issuance: July 23, 2010.

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

Amendment No.: 44.

Facility Operating License No. DPR–7: This amendment revises the License.

Date of initial notice in ***Federal Register***: June 15, 2010 (75 FR 33842).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 23, 2010.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 9, 2009.

Brief description of amendment: The amendment changes the frequency of control rod notch testing, as specified in Technical Specification (TS) surveillance requirement 4.1.3.1.2.a, from at least once per 7 days to at least once per 31 days. The amendment also adds the word "fully" to the Action for TS Limiting Condition for Operation 3.9.2 to clarify the requirement to fully insert all insertable control rods when the required source range monitor (SRM) instrumentation is inoperable. The proposed amendment is based on TS Task Force (TSTF) change, TSTF–475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action."

Date of issuance: July 21, 2010.

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

Amendment No.: 182.

Facility Operating License No. NPF–57: The amendment revised the TSs and the License.

Date of initial notice in ***Federal Register***: December 1, 2009 (74 FR 62836).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 2010.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company et al., Docket No. 52–011, Vogtle Electric Generating Plant ESP Site, Burke County, Georgia

Date of amendment request: May 24, 2010, as supplemented June 2 and 22, 2010.

Description of amendment request: This amendment revises the Vogtle Electric Generating Plant ESP Site Safety Analysis Report (SSAR) to change the classification of backfill over the slopes of the Units 3 and 4 excavations from Category 1 and 2 backfill to engineered granular backfill (EGB).

Date of issuance: July 9, 2010.

Effective date: As of date of issuance and shall be implemented within 15 days from the date of issuance.

Amendment No.: 3.

Early Site Permit No. ESP–004: Amendment revised the VEGP ESP SSAR.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. June 3, 2010 (75 FR 31477). The supplements dated June 2 and 22, 2010 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination. The June 3, 2010 notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The June 3, 2010 notice also provided an opportunity to request a hearing by August 2, 2010, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the requested amendment, state consultation, and final NSHC determination are contained in a safety evaluation dated July 9, 2010. The NRC staff prepared an environmental assessment (75 FR 39284) and determined that the requested amendment will not have a significant effect on the quality of the human environment.

Attorney for licensee: M. Stanford Blanton, Balch & Bingham, LLP.

NRC Branch Chief: Jeffrey Cruz.

Southern Nuclear Operating Company et al., Docket No. 52–011, Vogtle Electric Generating Plant ESP Site, Burke County, Georgia

Date of amendment request: April 20, 2010, as supplemented April 23 and 28, May 5, 10, 13, 20, and 24, 2010.

Description of amendment request: The amendment revised the Vogtle Electric Plant (VEGP) ESP Site Safety Analysis Report (SSAR) to allow the use of Category 1 and 2 backfill material from additional onsite areas that were not specifically identified in the VEGP ESP SSAR as backfill sources for the activities approved under the ESP and Limited Work Authorization.

Date of issuance: June 25, 2010.

Effective date: As of date of issuance and shall be implemented within 15 days from the date of issuance.

Amendment No.: 2.

Early Site Permit No. ESP–004: Amendment revised the VEGP ESP SSAR.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. May 6, 2010 (75 FR 24993). The supplements dated May 5, 10, 13, 20, and 24, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination. The May 6, 2010 notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The May 6, 2010 notice also provided an opportunity to request a hearing by July 6, 2010, but indicated that if the Commission makes a final

NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the requested amendment, state consultation, and final NSHC determination are contained in a safety evaluation dated June 25, 2010. The NRC staff prepared an environmental assessment (75 FR 36446) and determined that the requested amendment will not have a significant effect on the quality of the human environment.

Attorney for licensee: M. Stanford Blanton, Balch & Bingham, LLP.

NRC Branch Chief: Jeffrey Cruz.

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

Date of amendment request: October 10, 2010, as supplemented by letter dated March 8, 2010.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.1.7, "Rod Position Indication," TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N\Delta H$)," TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," and TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," for use of the Best Estimate Analyzer for Core Operations—Nuclear (BEACON) Power Distribution Monitoring System (PDMS), as described in WCAP–12472–P–A, "BEACON Core Monitoring and Operations Support System," to perform power distribution surveillances.

Date of issuance: July 23, 2010.

Effective date: As the date of issuance and shall be implemented by December 29, 2010.

Amendment No.: 188.

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

Date of initial notice in ***Federal Register***: January 26, 2010 (75 FR 4120). The supplemental letter dated March 8, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 23, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 30th day of July, 2010.

For The Nuclear Regulatory Commission.

Robert A. Nelson,

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

[FR Doc. 2010–19678 Filed 8–9–10; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2010–0002]

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of August 9, 16, 23, 30, and September 6, 13, 2010.

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

STATUS: Public and closed.

Week of August 9, 2010

Thursday, August 12, 2010

9:25 a.m. Affirmation Session (Public Meeting) (Tentative).

a. U.S. *Army Installation Command* (Schofield Barracks, Oahu, Hawaii, and Pohakuloa Training Area, Island of Hawaii, Hawaii), Appeal of Isaac D. Harp (Tentative).

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

9:30 a.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: Cindy Flannery, 301–415–0223).

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

Week of August 16, 2010—Tentative

There are no meetings scheduled for the week of August 16, 2010.

Week of August 23, 2010—Tentative

There are no meetings scheduled for the week of August 23, 2010.

Week of August 30, 2010—Tentative

There are no meetings scheduled for the week of August 30, 2010.

Week of September 6, 2010—Tentative

There are no meetings scheduled for the week of September 6, 2010.

Week of September 13, 2010—Tentative

There are no meetings scheduled for the week of September 13, 2010.

* 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: Rochelle Bavol, (301) 415–1651.

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

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 Angela Bolduc, Chief, Employee/Labor Relations and Work Life Branch, at 301–492–2230, TDD: 301–415–2100, or by e-mail at *angela.bolduc@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed electronically to 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), or send an e-mail to *darlene.wright@nrc.gov.*

Dated: August 5, 2010.

Rochelle C. Bavol,

Policy Coordinator, Office of the Secretary.

[FR Doc. 2010–19806 Filed 8–6–10; 4:15 pm]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NRC–2010–0274]

Final Regulatory Guide: Issuance, Availability

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of issuance and availability of Regulatory Guide, RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure."

FOR FURTHER INFORMATION CONTACT: Robert G. Roche, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, *telephone:* (301) 251–7645 or e-mail *Robert.Roche@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

The U.S. Nuclear Regulatory Commission (NRC or Commission) is issuing a new guide in the agency's "Regulatory Guide" series. This series was developed to describe and make available to the public information such as methods that are acceptable to the NRC staff for implementing specific