[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2010-0382]

[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 November 18, 2010, to December 1, 2010. The last biweekly notice was published on November 30, 2010 (75 FR 74091).

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, Public File Area O1F21, 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 O1-F21, 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 Electronic Information Exchange System, 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 1-866-672-7640. 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.*

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 amendment request: June 29, 2009, as supplemented June 24, 2010.

Description of amendment request: The proposed amendments would approve changes to the updated final safety analysis report to allow the use of fiber reinforce polymer on masonry walls for uniform pressure loads resulting from a tornado event.

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

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

   Response: Physical protection from a tornado event is a design basis criterion rather than a requirement of a previously analyzed [updated final safety analysis report] UFSAR accident analysis. The current licensing basis (CLB) for Oconee states that systems, structures, and components (SSC's) required to shut down and maintain the units in a shutdown condition will not fail as a result of damage caused by natural phenomena.

   The in-fill masonry walls to be strengthened using an FRP system are passive, non-structural elements. The use of an fiber reinforced polymer [FRP] system on existing Auxiliary Building masonry walls will allow them to resist uniform pressure loads resulting from a tornado and will not adversely affect the structure's ability to withstand other design basis events such as earthquakes or fires. Therefore, the proposed use of FRP on existing masonry walls will not significantly increase the probability or consequences of an accident previously evaluated.

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

Response: The final state of the FRP system is passive in nature and will not initiate or cause an accident. More generally, this understanding supports the conclusion that the potential for new or different kinds of accidents is not created.

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

Response: The application of an FRP system to existing Auxiliary Building masonry walls will act to enhance the margin of safety, e.g., the West Penetration Room walls, by increasing the walls' ability to resist tornado-induced differential pressure. Consequently, this change does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, Duke has concluded that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202.

NRC Branch Chief: Gloria Kulesa.

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 amendment request: July 14, 2010.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to adopt NRC Approved Technical Specification Task Force (TSTF) Change to the Standard TS, TSTF-52 concerning performance-based containment leakage testing requirements.

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

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

No. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits as delineated in [Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50] 10 CFR Part 50, Appendix J, Option B. The changes are consistent with current safety analyses. Although some of the proposed changes represent minor relaxation to existing [Technical Specifications] TS requirements, they are consistent with the requirements specified by Option B of 10 CFR Part 50, Appendix J. The systems affecting containment integrity related to this proposed amendment request are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed amendment. The proposed changes maintain an equivalent level of reliability and availability for all affected systems. In addition, maintaining leakage within analyzed limits assumed in accident analyses does not adversely affect either onsite or offsite dose consequences.

Therefore, adopting Appendix J, Option B does not significantly increase the probability or consequences of any accident previously evaluated.

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

No. No changes are being proposed which will introduce any physical changes to the existing plant design. The proposed changes are consistent with the current safety analyses. Some of the changes may involve revision in the testing of components; however, these are in accordance with the current safety analyses and provide for appropriate testing or surveillance that is consistent with 10 CFR Part 50, Appendix J, Option B. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current accident analyses. No new modes of operation are introduced by the proposed changes. The proposed changes maintain, at minimum, the present level of operability of any system that affects containment integrity.

Therefore, adoption of Appendix J, Option B will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

> No. The provisions specified in Option B of 10 CFR Part 50, Appendix J allow changes to Type B and Type C test intervals based upon the performance of past leak rate tests. 10 CFR Part 50, Appendix J, Option B allows longer intervals between leakage tests based on performance trends, but does not relax the leakage acceptance criteria. Changing test intervals from those currently provided in the TS to those provided in 10 CFR Part 50, Appendix J, Option B does not increase any risks above and beyond those that the [U S. Nuclear Regulatory Commission] NRC has deemed acceptable for the performance based option. In addition, there are risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. The proposed changes provide continued assurance of leakage integrity of containment without adversely affecting the public health and safety and will not significantly reduce existing safety margins.
>
> Therefore, adoption of Appendix J, option B 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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202.

NRC Branch Chief: Gloria Kulesa.

Duke Energy Carolinas, LLC, Docket Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station (ONS), Units 1, 2, and 3, Oconee County, South Carolina; Docket Nos. 50-369 and 50-370, McGuire Nuclear Station (MNS), Units 1 and 2, Mecklenburg County, North Carolina; Docket Nos. 50-413 and 50-414, Catawba Nuclear Station (CNS), Units 1 and 2, York County, South Carolina

Date of amendment request: September 16, 2010.

Description of amendment request: The proposed amendments would revise the Technical Specifications to update the qualification requirements for the Station Manager and Radiation

Protection Manager to meet or exceed the minimum qualifications in ANSI/ANS-3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants"

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

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

 Response: No

 The proposed change to [Technical Specifications] TS 5.3.1 is an administrative change to update the minimum qualification requirements for Station Manager and Radiation Protection Manager to meet or exceed ANSI/ANS 3.1-1993 as endorsed by Regulatory Guide 1.8, Revision 3, dated May 2000. This update for Station Manager and Radiation Protection Manager qualifications will also provide Oconee, McGuire, and Catawba the needed flexibility to appoint Station Managers and Radiation Protection Managers from a larger candidate pool. The current qualification requirements restrict the pool of personnel capable of performing the Station Manager and Radiation Protection Manager functions. This change will also revise the current Oconee, McGuire, and Catawba TS 5.3.1 qualification requirements for Station Manager and Radiation Protection Manager to be consistent among all three stations. The proposed change does not impact the physical configuration or function of plant structures, systems, or components or the manner in which structures, systems, or components are operated, maintained, modified, tested, or inspected. Updating the minimum qualification requirements for Station Manager and Radiation Protection Manager is not an initiator of any accident previously evaluated. Updating the minimum qualification requirements for Station Manager and Radiation Protection Manager is not an assumption in the consequence mitigation of any accident previously evaluated. Therefore, it is concluded that this 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

 The proposed change to TS 5.3.1 is an administrative change to update the minimum qualification requirements for Station Manager and Radiation Protection Manager to meet or exceed ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Revision 3, dated May 2000. This represents an update to current guidance. This update for Station Manager and Radiation Protection Manager qualifications will also provide Oconee, McGuire, and Catawba the needed flexibility to appoint Station Manager and Radiation Protection Manager from a larger candidate pool. The current qualification requirements restrict the

pool of personnel capable of performing the Station Manager and Radiation Protection Manager functions. This change will also revise the current Oconee, McGuire and Catawba TS 5.3.1 qualification requirements for Station Manager and Radiation Protection Manager to be consistent among all three stations.

The proposed change does not impact the physical configuration or function of plant structures, systems, or components or the manner in which structures, systems, or components are operated, maintained, modified, tested, or inspected. In addition, there is no change in the types or increases in the amounts of effluents that may be released offsite, and there is no increase in individual or cumulative occupational radiation exposure.

As the proposed change is administrative in nature, operation of the facility in accordance with the proposed amendment does 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 a margin of safety?

Response: No

The proposed change to TS 5.3.1 is an administrative change to update the minimum qualification requirements for Station Manager and Radiation Protection Manager to meet or exceed ANSI/ANS 3.1-1993 as endorsed by RG 1.8, Revision 3, day May 2000. This update for Station Manager and Radiation Protection Manager qualifications will, also provide Oconee, McGuire, and Catawba the needed flexibility to appoint Station Manager and Radiation Protection Manager from a larger candidate pool. The current qualification requirements restrict the pool of personnel capable of performing the Station Manager and Radiation Protection Manager functions. This change will also revise the current ONS, MNS, and CNS TS 5.3.1 qualification requirements for Station Manager and Radiation Protection Manager to be consistent among all three stations. The proposed change does not impact the physical configuration or function of plant structures, systems, or components or the manner in which structures, systems, or components 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. The proposed change is administrative in nature; thus operation of the facility in accordance with the proposed 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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202.

NRC Branch Chief: Gloria Kulesa.

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 amendment request: November 8, 2010.

Description of amendment request: The proposed amendments would approve revisions to the updated final safety analysis report to incorporate the licensee's reactor vessel internals inspection plan.

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

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

No. The proposed license amendment request provides the Reactor Vessel Internals Inspection Plan report. The report also provides a description of the inspection plan as it relates to the management of aging effects consistent with previous commitments. The inspection plan is based on MRP-227, Revision 0, "Pressurized Water Reactors Internals Inspection and Evaluation Guidelines" and describes using the ten Aging Management Program (AMP) elements in the current revision of NUREG-1801 "Generic Aging Lessons Learned" (GALL, Revision 1) report.

The inspection plan contains a discussion of the background of the Babcock and Wilcox designed plant Reactor Vessel Internals programs, first sponsored by the utilities through the Babcock and Wilcox Owner's Group and later by the Pressurized Water Reactor Owner's Group, culminating in a submittal to the Nuclear Regulatory Commission through the Electric Power Research Institute Materials Reliability Program. The inspection plan also contains a discussion of operational experience, time-limited aging analyses, and relevant existing programs.

The Reactor Vessel Internals Aging Management Program includes the inspection plan and demonstrates that the program adequately manages the effects of aging for Reactor Vessel Internals components and establishes the basis for providing reasonable assurance the Reactor Vessel Internals components will remain functional through the license renewal period of extended operation.

This license amendment request provides an inspection plan based on industry work and experiences as agreed to in Duke Energy's license renewal commitments for Reactor Vessel Internals Inspection. It is not an accident initiator; therefore, it will not increase the probability or consequences of an accident previously evaluated

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

No. The proposed Reactor Vessel Internals Inspection Plan does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The revised inspection plan is not an accident / event initiator. No new initiating events or transients result from the use of the Reactor Vessel Internals Inspection plan.

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

No. The proposed safety limits have been preserved. The License Amendment Request requests review and approval for the Reactor Vessel Internals Inspection plan that Duke Energy committed to provide prior to commencing inspections.

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.

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 amendment request: November 15, 2010.

Description of amendment request: The proposed amendments would approve changes to the updated final safety analysis report to allow operation of a reverse osmosis system during normal plant operation to remove silica from borated water storage tank and the spent fuel pool.

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?

No. The proposed change requests Nuclear Regulatory Commission (NRC) approval of design features and controls that will be used to ensure that periodic limited operation of a Reverse Osmosis (RO) System during Unit operation does not significantly impact the Borated Water Storage Tank (BWST) or Spent Fuel Pool (SFP) function or other plant equipment. Duke Energy evaluated the effect of potential failures, identified precautionary measures that must be taken before and during RO System operation, and required operator actions to protect affected structures, systems, and components (SSCs) important to safety.
The new high energy piping and non seismic piping being installed for the RO System is non QA-1 and is postulated to fail and cause an Auxiliary Building flood. Duke Energy determined that adequate time is available to isolate the flood source (BWST or SFP) prior to affecting SSCs important to safety.

The existing Auxiliary Building Flood evaluation postulates a single break in the nonseismic piping occurring in a seismic event. The addition of the RO System will not increase the frequency of a seismic event. This event does not consider the amount of non-seismic piping that is currently in the Auxiliary Building. The new piping is not more likely to fail as compared to the existing non-seismic piping. The existing postulated source of the pipe break in the Auxiliary Building is due to the piping not being seismically designed. The new RO System piping is considered a potential source of a single pipe break for the same reason. Since the accident itself is defined as the failure of non-seismic pipe, the new non-seismic piping does not increase the frequency of occurrence of an Auxiliary Building flood. The mitigation of an Auxiliary Building flood due to non seismic piping failure is by manual operator action. The same mitigation technique is used for the high energy line break.

The RO System takes suction from the top of the SFP to protect SFP inventory. Plant procedures will prohibit the use of the RO System during the time period directly after an outage that requires the Unit 1 & 2 SFP level to be maintained higher than the Technical Specification (TS) Limiting Condition for Operation (LCO) 3.7.11 level requirement. The higher level is required to support TS LCO 3.10.1 requirements for Standby Shutdown Facility (SSF) Reactor Coolant (RC) Makeup System operability (due to the additional decay heat from the recently offloaded spent fuel). Plant procedures will also specify the siphon be broken during this time period so the SFP water above the RO suction point cannot be siphoned off if the RO piping breaks. The proposed change does not impact the fuel assemblies, the movement of fuel, or the movement of fuel shipping casks. The SFP boron concentration, level, and temperature limits will not be outside of required parameters due to restrictions/requirements on the system's operation.

The BWST is used for mitigation of Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB) and Loss of Coolant Accidents (LOCAs). The SGTR and MSLB are bounded by the [small-break] SBLOCA analyses with respect to the performance requirements for the [high pressure injection] HPI System. In the normal mode of Unit operation, the BWST is not an accident initiator. The SFP is assumed to maintain acceptable criticality margin for all abnormal and accident conditions including Fuel Handling Accidents (FHAs) and cask drop accidents. Both the BWST and SFP are specified by TS requirements to have minimum levels/volumes and boron concentrations. The BWST also has TS requirements for temperature. Prior to RO operation, procedures will require that minimum required initial boron concentration, and initial level/volume be adjusted and that the RO System be operated for a specified maximum time period before readjusting volume and boron concentration prior to another RO session to ensure that the TS specified boron concentration and level/volume limits for both the SFP and the BWST are not exceeded during RO System operation. Thus, the design functions of the BWST and the SFP will continue to be met during RO System operation.

An Auxiliary Building flood due to a non-seismic RO System pipe break does not increase the consequences of the flood since the new non-seismic pipe break is bounded by the Auxiliary Building flood caused by existing non-seismic pipe breaks. Although the RO System will return water with lower boron concentration, procedural controls will ensure that the TS boron concentration level does not go below the limit. Thus, no adverse effects from decreased boron concentration levels will occur.

Since the BWST and SFP will still have TS required boron concentration and level/volume, the mitigation of a LOCA or FHA does not result in an increase in dose consequence.

Therefore, installation and operation of the RO System during Unit operation does not significantly increase the probability or consequences of any accident previously evaluated.

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

No. The RO System adds non-seismic piping in the Auxiliary Building. However, the break of a single non-seismic pipe in the Auxiliary Building has already been postulated as an event in the licensing basis. The RO System also does not create the possibility of a seismic event concurrent with a LOCA since a seismic event is a natural phenomena event. The RO System does not adversely affect the Reactor Coolant System pressure boundary. The suction to the RO System, when using the system for BWST purification, contains a normally closed manual seismic boundary valve so the seismic design criteria is met for separation of seismic/non-seismic piping boundaries.

Duke Energy also evaluated potential releases of radioactive liquid to the environment due to RO System piping failures. Design features and administrative controls preclude release of radioactive materials outside the Auxiliary Building. Releases inside the Auxiliary Building are bounded by existing analyses.

The SFP suction line is designed such that the SFP water level will not go below TS required levels, thus the fuel assemblies will have the TS required water level over them. Procedural controls will restrict the use of the RO System and require breaking vacuum on the SFP suction line when the SSF conditions require the SFP level be raised to support SSF RC Makeup System operability. Thus, the SFP water level will not be reduced below required water levels for these conditions. RO System operating restrictions will prevent reducing the SFP boron concentration below TS limits.

Therefore, operation of the RO System during Unit operation will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

No. The RO System adds non-seismic piping in the Auxiliary Building. Duke Energy evaluated the impact of RO System operation on SSCs important to safety and determined that procedural controls will ensure that TS limits for SFP and BWST volume, temperature and boron concentration will continue to be met during RO operation. For the BWST, these controls will ensure the TS minimum BWST boron concentration and level are available to mitigate the consequences of a

> small break LOCA or a large break LOCA. For the SFP, these controls ensure the assumptions of the fuel handling and cask drop accident analyses are preserved. Additionally, the failure of non seismic RO System piping will not significantly impact SSCs important to safety. The BWST level may drop below the TS required level due to a rupture of the non seismic piping during a seismic event. However, due to the low probability of a seismic event coupled with the relatively short period of time the RO System will be aligned to the BWST, the possibility of dropping below the TS required level does not involve a significant reduction in the margin of safety. In addition, Oconee's licensing basis does not assume a design basis event occurs simultaneously with a seismic event. The proposed change does not significantly impact the condition or performance of SSCs relied upon for accident mitigation. This change does not alter the existing TS allowable values or analytical limits. The existing operating margin between Unit conditions and actual Unit setpoints is not significantly reduced due to these changes. The assumptions and results in any safety analyses are not impacted. Therefore, operation of the RO System during Unit operation 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: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202.

NRC Branch Chief: Gloria Kulesa.

Energy Northwest, Docket No. 50-397, Columbia Generating Station, Benton County, Washington

Date of amendment request: September 30, 2010.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.1.7, “Standby Liquid Control (SLC) System,” to add Surveillance Requirement (SR) 3.1.7.9 to verify sodium pentaborate enrichment prior to the addition to the

SLC tank. The increase in boron-10 enrichment is needed to support future reloads of GE14 fuel by providing additional margin for preserving the shutdown objective of the SLC system. Reload analysis indicates that a core that is made up of a majority of GE14 fuel has a higher reactivity than previous Columbia Generating Station core designs warranting a corresponding increase in the shutdown capability of the SLC system.

<u>Basis for proposed no significant hazards consideration determination</u>: 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 SLC system is designed to provide sufficient negative reactivity to bring the reactor from full power to a subcritical condition at any time in a fuel cycle, without taking credit for control rod movement. The proposed changes to the SLC sodium pentaborate solution requirements maintain the capability of the SLC to perform this reactivity control function, and assure continued compliance with the requirements of 10 CFR 50.62 for ATWS [automatic transient without scram]. The proposed changes do not impact the LOCA [loss-of-coolant accident] suppression pool pH control function of SLC because single-pump minimum flow and sodium pentaborate solution concentration (weight percent) are not changed from the level credited in the LOCA analysis. The SLC is provided to mitigate ATWS events and LOCA and, as such, is not considered to be an initiator of the ATWS event, LOCA, or any other analyzed accident. The use of sodium pentaborate solution enriched with the boron-10 isotope, which is chemically and physically similar to the current solution, does not alter the design or operation of the SLC or increase the likelihood of a system malfunction that could increase the consequences of an accident.

   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.

Injection of sodium pentaborate solution into the reactor vessel has been considered in the plant design. The proposed changes revise the SLC boron solution requirements such that the capability of the SLC system to bring the reactor to a subcritical condition without taking credit for control rod movement is maintained, considering operation with an equilibrium core of GE14 fuel. The use of sodium pentaborate solution enriched with the boron-10 isotope, which is chemically and physically similar to the current solution, does not alter the design, function, or operation of the SLC system. The correct boron-10 enrichment is assured by the proposed addition of an SR to the TS. The solution concentration and volume are not changed; thus, the existing minimum volume and solution and piping temperature specified in the TS will ensure that the boron remains in solution and does not precipitate out in the SLC storage tank or in the SLC pump suction piping. The minimum volume and concentration specified in the TS ensure that the LOCA suppression pool pH control function is not impacted.

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 proposed changes revise the SLC boron solution requirements to maintain the capability of the SLC system to bring the reactor to a subcritical condition without taking credit for control rod movement. These changes support operation with an equilibrium core of GE14 fuel and assure continued compliance with the requirements of 10 CFR 50.62. The minimum required average boron-10 concentration in the reactor core, resulting from the injection of sodium pentaborate solution by the SLC system, has been determined using approved analytical methods. The analysis demonstrates that sufficient shutdown margin is maintained in the reactor such that the reactivity control function of the SLC system is assured. The additional quantity of boron included to account for imperfect mixing and leakage is maintained at 25 percent. No change in the solution pH or volume is made. Thus, the safety margin is maintained to bring the reactor subcritical in the event of an ATWS and to control suppression pool pH in the event of a LOCA.

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: William A. Horin, Esq., Winston & Strawn, 1700 K Street, N.W., Washington, D.C. 20006-3817.

NRC Branch Chief: Michael T. Markley.

Entergy Nuclear Operations, Inc., Docket No. 50-255, Palisades Nuclear Plant, Van Buren County, Michigan

Date of amendment request: July 20, 2010.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.8.3, “Diesel Fuel, 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 they may be modified under licensee control. The TS are modified so that the stored diesel fuel oil and lube oil inventory will require that a 7-day supply be available for either diesel generator. Condition A and Condition B in the Action table are revised and Surveillance Requirements (SR) 3.8.3.1 and 3.8.3.2 are revised to reflect the above change.

The proposed changes also revise TS 3.8.3 by reducing the Completion Time for Condition C. Condition C currently requires that an inoperable fuel transfer system associated with fuel oil transfer pump P-18A be restored to operable status within 15 hours. The proposed TS change reduces the Completion Time for this Required Action from 15 to 12 hours. The Completion Time is reduced to reflect the amount of time that an emergency diesel generator fuel oil day tank can support emergency diesel generator operation under design conditions.

<u>Basis for proposed no significant hazards consideration determination</u>: 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 emergency diesel generators, and the volume equivalent to a 6-day supply, to licensee control. The specific volume of fuel oil equivalent to a 7-day and 6-day supply is calculated using the NRC approved methodology described in Regulatory Guide 1.137, Revision 1, “Fuel Oil Systems for Standby Emergency diesel generators” and ANSI N195-1976, "Fuel Oil Systems for Standby Diesel Generators." The specific volume of lube oil equivalent to a 7-day and 6-day supply is based on the emergency diesel generator manufacturer's consumption values for the run time of the diesel generator. 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 supply have not changed, neither the probability or the consequences of any accident previously evaluated will be affected.

   The proposed change also reduces the Completion Time for TS 3.8.3 Condition C for an inoperable P-18A fuel transfer system from 15 hours to 12 hours. Reducing the Completion Time to 12 hours bounds the 13.5-hour time duration that the emergency diesel generator day tank will support emergency diesel generator operation under accident loading conditions. The change in Completion Time does not affect required TS actions if the Completion Time is exceeded. The Completion Time change does not affect the probability or consequences of an accident previously evaluated.

   Therefore, 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.

   The proposed fuel oil and lube oil 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. The change does not alter assumptions made in the safety analysis but ensures that the

emergency diesel generator operates as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

The proposed change also reduces the Completion Time for TS 3.8.3 Condition C for an inoperable P-18A fuel transfer system from 15 hours to 12 hours. This change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). This change does not create a condition in which a new or different kind of accident can occur. It does not alter assumptions made in the 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 change involve a significant reduction in a margin of safety?

Response: No.

The proposed change relocates the volume of fuel oil and lube oil required to support 7-day operation of either emergency diesel generator, and the volume equivalent to a 6-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.

The proposed change also reduces the Completion Time for TS 3.8.3 Condition C for an inoperable P-18A fuel transfer system from 15 hours to 12 hours. There are no adverse affects on margins of safety since a more stringent operability requirement will be applied to the P-18A fuel transfer system.

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

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

Attorney for licensee: Mr. William Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Ave., White Plains, NY 10601.

NRC Branch Chief: Robert J. Pascarelli.

<u>Exelon Generation Company, LLC, and PSEG Nuclear, LLC, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, York and Lancaster Counties, Pennsylvania.</u>

<u>Date of amendment request</u>: March 24, 2010, as supplemented by letter dated July 23, 2010.

<u>Description of amendment request</u>: The proposed amendment would revise Technical Specification (TS) Section 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time for Condition C (i.e., two SLC subsystems inoperable for reasons other than Condition A) from 8 hours to 72 hours.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC), 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

The proposed amendment revises Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to extend the completion time (CT) for Condition C (i.e., "Two SLC subsystems inoperable for reasons other than Condition A.") from eight hours to 72 hours.

The proposed change is based on a risk-informed evaluation performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications."

The proposed amendment modifies an existing CT for a dual-train SLC System inoperability. The condition evaluated, the action requirements, and the associated CT do not impact any initiating conditions for any accident previously evaluated.

The proposed amendment does not increase postulated frequencies or the analyzed consequences of an Anticipated Transient Without Scram (ATWS). Requirements associated with 10 CFR 50.62 will continue to be met. In addition, the proposed amendment does not increase postulated frequencies or the analyzed consequences of a large-break loss-of-coolant accident for which the SLC System is used for pH control. The new action requirement provides appropriate remedial actions to be taken in

response to a dual-train SLC System inoperability while minimizing the risk associated with continued operation. 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.

The proposed amendment revises TS 3.1.7 to extend the CT for Condition C from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. This proposed TS amendment does not change the design function of the SLC System and does not affect the system's ability to perform its design function. The SLC System provides a method to bring the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. Required actions and surveillance requirements are sufficient to ensure that the SLC System functions are maintained. No new accident initiators are introduced by this amendment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed amendment revises TS 3.1.7 to extend the CT for Condition C from eight hours to 72 hours. The proposed amendment does not involve any change to plant equipment or system design functions. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated and the setpoints for the actuation of equipment relied upon to respond to an event.

Safety margins applicable to the SLC System include pump capacity, boron concentration, boron enrichment, and system response timing. The proposed amendment does not modify these safety margins or the setpoints at which SLC is initiated, nor does it affect the system's ability to perform its design function. In addition, the proposed change complies with the intent of the defense-in-depth philosophy and the principle that sufficient safety margins are maintained consistent with RG 1.177 requirements (i.e., Section C, "Regulatory Position," paragraph 2.2, "Traditional Engineering Considerations"). Therefore, the proposed 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves NSHC.

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

NRC Branch Chief: Harold K. Chernoff.

NextEra Energy Duane Arnold, LLC, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: August 12, 2010.

Description of amendment request: A change is proposed to the technical specifications to allow a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an unavailable barrier, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). LCO 3.0.9 will be added to individual TS providing this allowance.

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 by affirming the applicability of the model analysis presented in the *Federal Register* notice dated October 3, 2006, starting on page 71 FR 58452, which is presented below:

> *Criterion 1: The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*
>
> The proposed change allows a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an unavailable barrier if risk is assessed and managed. The postulated initiating events which may require a functional barrier are limited to those with low frequencies of occurrence, and the overall TS system safety function would still be available for the majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on the allowance provided by proposed LCO 3.0.9 are no different than the consequences of an accident while relying on the TS required

actions in effect without the allowance provided by proposed LCO 3.0.9. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns.

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering supported system TS when inoperability is due solely to an unavailable barrier, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns.

Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an unavailable barrier, if risk is assessed and managed. The postulated initiating events which may require a functional barrier are limited to those with low frequencies of occurrence, and the overall TS system safety function would still be available for the majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in [Regulatory Guide] RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.9 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant as indicated by the anticipated low levels of associated risk (ICCDP and ICLERP) as shown in Table 1 of Section 3.1.1 in the [model] Safety Evaluation [on page 71 FR 58450 of the *Federal Register* dated October 3, 2006].

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

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

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

NRC Branch Chief: Robert J. Pascarelli.

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

Date of amendment request: November 11, 2010.

Description of Amendment Request: The licensee proposes to amend the operating license for Virgil C. Summer Nuclear Station (VCSNS), by revising the Technical Specifications (TS) and SCE&G proposes to provide surveillance enhancements that will improve operation and testing of the Emergency Diesel Generators (EDG). The changes will provide a more restrictive voltage and frequency band for operation when not connected in parallel with the offsite sources.

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

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

No.

The changes proposed by this license amendment will revise the Surveillance Requirements of Technical Specification 3/4.8.1, AC SOURCES - OPERATING, to expand the continuous rated load specification to a range of 90% to 100% of the continuous rated load, specify an overload range of 105% to 110% of the

continuous rated load, add a power factor limit while testing, allow gradual loading and unloading of the EDG, specify a maximum frequency for the overspeed limit, specify a maximum allowable overspeed voltage, and add a more restrictive voltage and frequency band for testing during steady state operation.

The majority of these changes are being proposed in order to implement recommendations contained in [Institute of Nuclear Power Operations] INPO Significant Operating Experience Report (SOER) 03-01, *Emergency Power Reliability*, Recommendation Number 5, which recommends that the utility review testing practices for emergency power systems to verify that the practices are representative of actual demand conditions and appropriately exercise equipment that is expected to respond in an actual demand condition. These changes are based on the guidance provided by Regulatory Guide 1.9, Revision 3, *Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plant*.

The more restrictive voltage and frequency band for testing during steady state operation is proposed to ease the impact of EDG voltage and frequency that are being incorporated into the Charging Pump performance requirements. The allowable voltage and frequency uncertainty limits for steady state operation are being reduced. This will ensure that the Charging Pumps continue to operate within their analyzed range.

These changes do not affect the probability or consequences of an accident previously evaluated because the proposed changes do not make a change to any accident initiator, initiating condition, or assumption. The proposed changes do not involve a significant change to the plant design or operation. These changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path or alter a potential radiological release. 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?

No.

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

No changes are being made in equipment hardware or software, operational philosophy, testing frequency, or how the system actually operates. Therefore,

the proposed amendment will 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 a margin of safety?

No.

These changes do not involve a significant reduction in a margin of safety because the proposed changes do not reduce the margin of safety that exists in the present Technical Specifications or Updated Final Safety Analysis Report. The operability requirements of the Technical Specifications are consistent with the initial condition assumptions of the safety analyses. The proposed changes do not affect the Action statement requirements for the various levels of degradation in the EDG. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: J. Hagood Hamilton, Jr., South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Branch Chief: Gloria Kulesa.

Southern Nuclear Operating Company, Inc. (SNC), Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Houston County, Alabama

Date of amendment request: October 29, 2010.

Description of amendment request: The proposed amendments request the adoption of an approved change to the standard technical specifications for Westinghouse Plants (NUREG-

1431), to allow relocation of specific Technical Specifications (TS) surveillance frequencies to a licensee-controlled program. The proposed change is described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3, ''Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b'' (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080280275), and was described in the Notice of Availability published in the *Federal Register* (FR) on July 6, 2009 (74 FR 31996). The proposed changes are consistent with NRC-approved TSTF-425, Revision 3. The proposed change relocates surveillance frequencies to a licensee-controlled program, the surveillance frequency control program. This change is applicable to licensees using probabilistic risk guidelines contained in NRC-approved [Nuclear Energy Institute] NEI 04-10, ''Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies,'' (ADAMS Accession No. 071360456).

The licensee affirmed the applicability to the FNP of the model no significant hazards consideration determination provided in the FR on July 6, 2009 (74 FR 31996), in its application dated October 29, 2010.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the analysis of the issue of no significant hazards consideration 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 specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program [SFCP]. 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 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.

   No new or different accidents result from utilizing the proposed change. 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. Does the proposed change involve a significant reduction in the 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, the licensee 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.

Based upon the reasoning presented above, licensee concludes that the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of Amendment.

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. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Branch Chief: Gloria J. Kulesa.

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

Date of amendment request: October 29, 2010.

Description of amendment request: The proposed amendments request the adoption of an approved change to the standard technical specifications for General Electric Plants, BWR/4 (NUREG-1433), to allow relocation of specific Technical Specification (TS) surveillance frequencies to a licensee-controlled program. The proposed change is described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3, ''Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b.'' (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080280275), and was described in the Notice of Availability published in the *Federal Register* (FR) on July 6, 2009 (74 FR 31996). The proposed changes are consistent with NRC-approved TSTF-425, Revision 3. The proposed change relocates surveillance frequencies to a licensee-controlled program, the surveillance frequency control program. This change is applicable to licensees using probabilistic risk guidelines contained in NRC-approved [Nuclear Energy Institute] NEI 04-10, ''Risk-Informed

Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies,'' (ADAMS Accession No. 071360456). The licensee affirmed the applicability to the HNP of the model no significant hazards consideration determination provided in the FR on July 6, 2009 (74 FR 31996) in its application dated October 29, 2010.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the analysis of the issue of no significant hazards consideration 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 specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program [SFCP]. 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 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.

   No new or different accidents result from utilizing the proposed change. 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. Does the proposed change involve a significant reduction in the 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, SNC 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.

   Based upon the reasoning presented above, licensee concludes that the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of Amendment.

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. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Branch Chief: Gloria J. Kulesa.

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: February 18, 2010, as supplemented on November 12, 2010 (TS-468).

<u>Description of amendment request</u>: The proposed amendment would modify Technical Specification 3.8.1 to extend the completion time (CT) for the return of an inoperable emergency diesel generator (DGs) to operable status from 7 days to 14 days, based on the availability of two non-safety related temporary diesel generators (TDGs). Commensurate changes to the maximum completion times were also proposed, extending the times from 14 to 21 days in Required Actions A.3 and B.4. The change also eliminates a historical footnote for a previous CT for Unit 3 only that is no longer needed.

<u>Basis for proposed no significant hazards consideration determination</u>: 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
   The proposed changes do not affect the design of the DGs, the operational characteristics or function of the DGs, the interfaces between the DGs and other plant systems, or the reliability of the DGs. Required Actions and their associated CTs are not considered initiating conditions for any UFSAR [updated final safety analysis report] accident previously evaluated, nor are the DGs considered initiators of any previously evaluated accidents. The DGs are provided to mitigate the consequences of previously evaluated accidents, including a loss of off-site power.

   The consequences of previously evaluated accidents will not be significantly affected by the extended DG CT, because a sufficient number of onsite Alternating Current [AC] power sources will continue to remain available to perform the accident mitigation functions associated with the DGs, as assumed in the accident analyses. In addition, as a risk mitigation and defense-in-depth action, an independent AC power source, via two available TDGs, will be available to support the ESF [engineered safety feature] bus with the inoperable DG during a SBO [station blackout].

   Therefore, the proposed changes do 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

The proposed change does not involve a change in the permanent design, configuration, or method of operation of the plant. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. The proposed changes allow operation of the unit to continue while a DG is repaired and retested with the TDGs in standby to mitigate a SBO event. The proposed extensions do not affect the interaction of a DG with any system whose failure or malfunction can initiate an accident. As such, no new failure modes are being introduced. 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 a margin of safety?

Response: No

The proposed changes do not alter the permanent plant design, including instrument set points, nor does it change the assumptions contained in the safety analyses. The standby TDG alternate AC system is designed with sufficient redundancy such that a DG may be removed from service for maintenance or testing. The remaining seven DGs are capable of carrying sufficient electrical loads to satisfy the UFSAR requirements for accident mitigation or unit safe shutdown. The proposed changes do not impact the redundancy or availability requirements of offsite power supplies or change the ability of the plant to cope with station blackout events. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, 6A West Tower, Knoxville, Tennessee 37902.

NRC Branch Chief: Douglas A. Broaddus.

mitigation or unit safe shutdown. The proposed changes do not impact the redundancy or availability requirements of offsite power supplies or change the ability of the plant to cope with station blackout events. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, 6A West Tower, Knoxville, Tennessee 37902.

NRC Branch Chief: Douglas A. Broaddus.

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

Date of amendment request: October 21, 2010.

Description of amendment request: The proposed amendment would correct a typographical error in Section 5, Administrative Controls, of the Technical Specifications (TSs). The current TSs, on page 5.0-31, has two paragraphs numbered as 5.7.2d.3. The amendment proposes to renumber the second paragraph as 5.7.2d.4. The typographical error was introduced in Amendment No. 123 issued on March 31, 1999.

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

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

   Response: No.

   The proposed change is administrative in nature. The change involves correcting a typographical error. This change does not affect possible initiating events for accidents previously evaluated or alter the

configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the TS remain unchanged.

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 previously evaluated?

   Response: No.

   The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change.

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

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

   Response: No.

   The proposed change is administrative in nature. Since there [are] no changes to the operation of the facility or the physical design, the Updated Safety Analysis Report (USAR) design basis, accident assumptions, or TS Bases are not affected.

   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: Jay Silberg, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, DC 20037.

NRC Branch Chief: Michael T. Markley.

PREVIOUSLY PUBLISHED NOTICES OF

CONSIDERATION OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES, PROPOSED NO

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION,

AND OPPORTUNITY FOR A HEARING

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

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

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

Date of application for amendment: September 8, 2010.

Brief description of amendment: The licensee proposed to delete the Technical Specification requirements related to the containment hydrogen recombiners and the hydrogen monitors, in accordance with Nuclear Energy Institute Technical Specification Task Force (TSTF) initiative designated as TSTF-447.

Date of publication of individual notice in FEDERAL REGISTER: October 14, 2010 (75 FR 63209).

Expiration date of individual notice: December 13, 2010.

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 email to *pdr.resource@nrc.gov*.

<u>Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, Maricopa County, Arizona</u>

<u>Date of application for amendment</u>: November 30, 2009, as supplemented by letter dated July 22, 2010.

<u>Brief description of amendment</u>: The amendments revised Table 3.3.5-1 of Technical Specification (TS) 3.3.5, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," to raise the refueling water tank (RWT) low level allowable values for the recirculation actuation signal; raised the minimum required RWT volume shown in TS Figure 3.5.5-1 of TS 3.5.5, "Refueling Water Tank (RWT)"; and implemented a time-critical operator action to close the RWT isolation valves, including consideration of a potentially more limiting single failure of a low-pressure safety injection pump to automatically stop, as designed, on an recirculation actuation signal.

<u>Date of issuance</u>: November 24, 2010.

<u>Effective date</u>: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

<u>Amendment No</u>.: Unit 1 - 182; Unit 2 - 182; Unit 3 - 182.

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendment revised the Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: April 20, 2010 (75 FR 20629). The supplemental letter dated July 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*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 24, 2010.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, LLC, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: April 5, 2010

Brief description of amendments: The amendment made title changes and corrections within Technical Specification (TS) 5.0, "Administrative Controls." Specifically, the changes included:

(1) Replacement of the use of plant specific titles to generic titles consistent with TS Task Force (TSTF) Traveler TSTF-65, Revision 1, "Use of Generic Titles for Utility Positions,"

(2) Changes made to more closely align selected TSs with the Improved Standard TSs, and

(3) Administrative changes to specified TSs.

Date of issuance: November 22, 2010.

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

Amendment Nos.: 296 for Unit 1 and 272 for Unit 2.

Renewed Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: June 1, 2010 (75 FR 30443).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated November 22, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of December 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation