[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2009-0363]

BIWEEKLY NOTICE

APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 30, 2009 to August 12, 2009. The last biweekly notice was published on August 11, 2009 (74 FR 40233).

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, Rulemaking and Directives Branch (RDB), 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 RDB 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, Public File Area O1F21, 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 petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner/requestor to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, 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, which the NRC promulgated in August 2007 (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 petitioner/requestor should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by calling (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/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate is available on NRC's public website at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it 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 filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE 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 EIE 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 through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC electronic filing Help Desk, which is available between 8:00 a.m. and 8:00 p.m., Eastern Time, Monday through Friday, excluding government holidays. The toll-free help line number is 1-866-672-7640. A person filing electronically may also seek assistance by sending an email to the NRC electronic filing Help Desk at MSHD.Resource@nrc.gov.

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.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the request and/or petition should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)-(viii).

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, an Atomic Safety and Licensing Board, or a 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 submissions.

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, Public File Area O1F21, 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 email to pdr.resource@nrc.gov.

Entergy Gulf States Louisiana, LLC, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: June 29, 2009.

Description of amendment request: The proposed amendment would revise the requirements in Technical Specification (TS) 5.5.6, "Inservice Testing Program." TS 5.5.6 currently contains references to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI as the source of requirements for the inservice testing (IST) of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would delete the references to Section XI of the ASME Code and incorporate references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code). In addition, the proposed amendment would address the applicability of Surveillance Requirement 3.0.2 to other normal and accelerated frequencies as 2 years or less in the IST program. These changes are consistent with changes identified in the Improved Standard Technical Specifications (ISTS) by Technical Specification Task Force Traveler (TSTF) Nos. 479 and 497.

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

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

   Response: No.

   The proposed change revises the Technical Specification Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

   The proposed changes revise TS 5.5.6 for RBS to conform to the requirements of 10 CFR 50.55a(f) regarding the IST of pumps and valves for the third 10-Year Interval. The current TS reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the IST of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code instead. This is consistent with 10 CFR 50.55a(f). The proposed changes are administrative in nature.

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

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

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

   Response: No.

   The proposed change revises the Technical Specification Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

   The proposed TS changes do not involve physical changes to the facility. In addition, the proposed changes have no affect on plant configuration, or method of operation of plant structures, systems, or components.

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

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

   Response: No.

   The proposed change revises the Technical Specification Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

   The change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The IST of the Class 1, 2, and 3 pumps and valves continue to meet the appropriate requirements.

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

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

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

NRC Branch Chief: Michael T. Markley.

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment request: July 3, 2009.

Description of amendment request: The proposed amendments would revise the operability requirements and actions in Technical Specification (TS) 3.4.15, “RCS [Reactor Coolant System] Leakage Detection Instrumentation,” and the associated Bases Section to reflect the revised TSs.

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

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

   Response: No.

   The proposed change reduces the time allowed for the plant to operate when the only Technical Specification (TS) 3.4.15 operable Reactor Coolant System (RCS) leakage instrumentation monitor is the containment atmosphere gaseous radioactivity monitor, and revises the basis for operability for the containment sump monitors, containment atmosphere particulate radioactivity monitor, containment atmosphere gaseous radioactivity monitor, and the containment fan cooler unit condensate collection monitor. The proposed change increases the allowed operating time when all RCS leakage detection system instrumentation is inoperable. The proposed change also removes the word “required” from TS 3.4.15 Condition A, Required Action A.2, Condition B, and Required Action B.2, revises TS 3.4.15 Condition A to apply to any containment sump monitor, and revises the name of the containment fan cooler unit (CFCU) condensate collection monitor in the TS 3.4.15 Actions. The monitoring of RCS leakage is not a precursor to any accident previously evaluated. The monitoring of RCS leakage is not used to mitigate the consequences of any accident previously evaluated.

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 accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve a physical alteration of the plant or the addition of new or different type of equipment. The change does not involve a change in how the plant is operated.

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

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

   Response: No.

   The change that reduces the allowed time of operation with only the least accurate containment atmosphere gaseous radiation monitor operable increases the margin of safety by increasing the likelihood that an increase in RCS leakage will be detected before it potentially results in gross failure. For the change that allows a limited period of time to restore at least one RCS leakage detection monitor to operable status when all leakage detection monitors are inoperable, two sources of diverse leakage detection capability are required to be provided during the limited period. Allowing a limited period of time to restore at least one RCS leakage detection instrument to operable status before requiring a plant shutdown avoids the situation of putting the plant through a thermal transient without RCS leakage monitoring. The change to TS 3.4.15 Condition A, Required Action A.2, Condition B, Required Action B.2, Condition C, and Required Action C.2.2 is consistent with TS [Limiting Condition for Operation] 3.4.15 and does not impact the RCS leakage instrumentation. The revision to the TS bases for operability of the RCS leakage instrumentation monitors does not involve a change in the

> leakage instrumentation and is consistent with the original design of the leakage instrumentation.
>
> 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 requests involve no significant hazards consideration.

Attorney for licensee: Jennifer Post, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Branch Chief: Michael T. Markley.

PSEG Nuclear LLC, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: April 9, 2009.

Description of amendment request: The proposed amendment would relocate Technical Specification (TS) requirements pertaining to communications during refueling operations (TS 3/4.9.5), manipulator crane operability (TS 3/4.9.6), and crane travel (TS 3/4.9.7) to the Technical Requirements Manual.

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review 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 would relocate TS requirements to the Technical Requirements Manual (TRM) which is a licensee-controlled document. The TS requirements to be relocated relate to control room communications during refueling, operability of the manipulator crane and auxiliary hoist for movement of control rods or fuel assemblies within the reactor pressure vessel, and control of heavy loads over fuel assemblies in the fuel storage pool. Once relocated, any future changes would be controlled by 10 CFR 50.59. The proposed amendment is administrative in nature from the standpoint that the current TS requirements would be relocated verbatim to the TRM. There are no physical plant modifications associated with this change. The proposed amendment would not alter the way any structure, system, or component (SSC) functions and would not alter the way the plant is operated. As such, the proposed amendment would have no impact on the ability of the affected SSCs to either preclude or mitigate an accident. 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 would not change the design function or operation of the SSCs involved and would not impact the way the plant is operated. As such, the proposed change would not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

The margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation to the public. There are no physical plant modifications associated with the proposed amendment. The proposed amendment would not alter the way any SSC functions and would not alter the way the plant is operated. The proposed amendment would not introduce any new uncertainties or change any existing uncertainties associated with any safety limit. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the NRC staff concludes that the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Jeffrie J. Keenan, Esquire, Nuclear Business Unit - N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

<u>NRC Branch Chief</u>: Harold K. Chernoff.

<u>STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas</u>

<u>Date of amendment request</u>: February 3, 2009.

<u>Description of amendment request</u>: The proposed amendment would revise the Operating Licenses to deviate from certain South Texas Project Fire Protection Program requirements. The amendment will allow the performance of operator manual actions to achieve and maintain safe shutdown in the event of a fire in lieu of meeting circuit separation protection requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R, Section III.G.2 for Fire Area 31.

<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 design functions of structures, systems and component[s] are not impacted by the proposed change. The proposed change involves operator manual actions in response to a fire and will not initiate an event. The proposed actions do not increase the probability of occurrence of a fire or any other accident previously evaluated.

   The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fire. No significant consequences result from the performance of the proposed actions.

   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 design functions of structures, systems and component[s] are not impacted by the proposed amendment. The proposed change involves operator manual actions in response to a fire. They do not involve new failure mechanisms or malfunctions that can initiate a new accident.

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

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

   Response: No.

   Adequate time is available to perform the proposed operator manual actions to account for uncertainties in estimates of the time available and in estimates of how long it takes to diagnose and execute the actions. The actions are straightforward and do not create any significant concerns. The actions have been verified that they can be performed through demonstration and they are proceduralized. The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fire.

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

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: March 3, 2009.

Description of amendment request: The proposed change would revise the Operating Licenses to deviate from certain South Texas Project Fire Protection Program requirements. The amendment will allow the performance of operator manual actions to achieve and maintain safe shutdown in the event of a fire in lieu of meeting circuit separation protection requirements of

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R, Section III.G.2 for Fire Area 27.

<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 design functions of structures, systems and components are not impacted by the proposed change. The proposed change involves operator manual actions in response to a fire, and will not initiate an event. The proposed actions do not increase the probability of occurrence of a fire or any other accident previously evaluated.

   The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fire. No significant consequences result from the performance of the proposed actions.

   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 design functions of structures, systems and components are not impacted by the proposed amendment. The proposed change involves operator manual actions in response to a fire. They do not involve new failure mechanisms or malfunctions that can initiate a new accident.

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

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

   Response: No.

   Adequate time is available to perform the proposed operator manual actions to account for uncertainties in estimates of the time available and

> in estimates of how long it takes to diagnose and execute the actions. The actions are straightforward and do not create any significant concerns. The actions have been verified that they can be performed through demonstration and they are proceduralized. The proposed actions are feasible and reliable and demonstrate that the unit can be safely shutdown in the event of a fare.
>
> 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendment involves no significant hazards consideration.

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: July 9, 2009.

Description of amendment request: The proposed amendment would allow use of a dedicated on-line core power distribution monitoring system (PDMS) to enhance surveillance of core thermal limits and would revise Technical Specification (TS) TS 1.1, “Definitions,” TS 3.1.8, “Rod Position Indication,” TS 3.2.1, “Heat Flux Hot Channel Factor,” TS 3.2.4, “Quadrant Power Tilt Ratio (QPTR)”, and TS 3.3.1, “Reactor Trip System (RTS) Instrumentation.”

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 Power Distribution Monitoring System (PDMS) performs essentially continuous core power distribution monitoring with data input from existing plant instrumentation. This system utilizes an NRC-approved Westinghouse proprietary computer code, i.e. Best Estimate Analyzer for Core Operations - Nuclear (BEACON), to provide data reduction for incore flux maps, core parameter analysis, load follow, operation simulation, and core prediction. The PDMS does not provide any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will, therefore, not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

   Use of the PDMS supports maintaining the core power distribution within required limits. Further, continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This result in more time (i.e. earlier determination of an adverse condition developing) for operator action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

   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.

   Other than use of the PDMS to monitor core power distribution, implementation of the PDMS and associated Technical Specification changes has no impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with implementation of the PDMS do not result in a change to the design basis of any plant component or system. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met.

The proposed changes do not result in any event previously deemed incredible being made credible. Implementation of the PDMS will not result in any additional adverse condition and will not result in any increase in the challenges to safety systems. The cycle specific variables required by the PDMS are calculated using NRC approved methods. The Technical Specifications will continue to require operation within the required core operating limits, and appropriate actions will continue to be taken when or if limits are exceeded.

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

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

Response: No.

No margin of safety is adversely affected by the implementation of the PDMS. The margins of safety provided by current Technical Specification requirements and limits remain unchanged, as the Technical Specifications will continue to require operation within the core limits that are based on NRC approved reload design methodologies. Appropriate measures exist to control the values of these cycle specific limits, and appropriate actions will continue to be specified and taken for when limits are violated. Such actions remain unchanged.

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.

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

<u>NRC Branch Chief</u>: L. Raghavan.

<u>Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri</u>

<u>Date of amendment request</u>: March 20, 2009.

Description of amendment request: The proposed amendment would revise the Operating License No. NPF-30 for Callaway Plant, Unit 1, in order to incorporate a change to Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," which establishes the program for leakage rate testing of the containment, as required by Title 10 of *Code of Federal Regulations* (10 CFR) Section 50.54, "Conditions of licenses," Subsection (o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance Based Requirements," as modified by approved exemptions. Specifically, the TS 5.5.16 would be revised to reflect a one-time 5-year deferral of the containment Type A integrated leak rate test (ILRT) from once in 10 years to once in 15 years.

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

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

   Response: No

   The proposed change will revise Callaway Plant TS 5.5.16, "Containment Leakage Rate Testing Program," to reflect a one-time, five-year extension for the containment Type A test date to enable the implementation of a 15-year test interval. While the containment is designed to contain radioactive material that may be released from the reactor core following a design basis Loss-of-Coolant Accident (LOCA), the test interval associated with Type A testing is part of ensuring the plant's ability to mitigate the consequences of accidents described in the FSAR [Final Safety Analysis Report] and does not involve a precursor or initiator of any accident previously evaluated. Thus, the proposed change to the Type A test interval cannot increase the probability of an accident previously evaluated in the FSAR.

   Type A testing does provide assurance that the containment will not exceed allowable leakage rate criteria specified in the TS and will continue to perform its design function following an accident. However, per NUREG-1493, "Performance-Based Containment Leak-Test Program," Type A tests identify only a few potential leakage paths that cannot be identified by Type B and C testing. The current Type B and C penetration test frequencies for Callaway are established based on performance, using the requirements of 10 CFR 50, Appendix J, Option B, and the Type B and C testing requirements will not be changed as a result of the proposed license amendment. As a result, with respect to the consequences of an accident, a risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

   Response: No

   The proposed change is for a one-time, five-year extension of the Type A test for Callaway Plant and will not affect the control parameters governing unit operation or the response of plant equipment to transient or accident conditions. The proposed change does not introduce new equipment, modes of system operation, or failure mechanisms.

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

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

   Response: No

   The Callaway Plant containment consists of the concrete containment building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

   The containment is a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome and a reinforced concrete base slab. The inside structure is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. A post-tensioning system is used to prestress the cylindrical shell and dome.

The concrete containment building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage-limiting boundary of the containment. Maintaining operability of the containment will limit leakage of fission product radioactivity released from the containment to the environment.

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by an ILRT, as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The existing 10-year interval at Callaway Plant is based on past performance. Previous Type A tests conducted at Callaway Plant indicate that leakage from containment has been less than all 10 CFR 50 Appendix J, Option B, leakage limits.

The proposed change for a one-time extension of the Type A test does not affect the method for Type A, B, or C testing or the test acceptance criteria. Type B and C testing will continue to be performed at the frequency required by Callaway Plant Technical Specifications. The containment inspections that are performed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection," and 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provide a high degree of a assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

In NUREG-1493, "Performance-Based Containment Leak-Test Program," the NRC indicated that a 20-year extension for Type A testing resulted in an imperceptible increase in risk to the public. The NUREG-1493 study also concluded that, generically, the design containment leak rate contributes a very small amount to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk. AmerenUE has conducted risk assessments to determine the impact of a one-time change to the Callaway Plant Type A test schedule from a baseline value of once in 10 years to once in 15 years for the risk measures of Large Early Release Frequency (LERF), Total Population Dose, and Conditional Containment Failure Probability (CCFP). The results of the risk assessments indicate that the proposed change to the Callaway Plant Type A test schedule has a minimal impact on public risk.

Based on the above and on previous Type A test results for the Callaway Plant containment, the current containment surveillance program, and the results of the AmerenUE risk assessment, there is no reduction in the effectiveness of the Callaway Plant containment as a barrier to the

> release of the post-accident containment atmosphere to the public or to personnel in the Control Room. Thus, 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: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: May 4, 2009.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.7.3, "Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs), and Main Feedwater Regulating Valve Bypass Valves (MFRVBVs)," so that the limiting condition for operation (LCO) and Applicability more accurately reflect the conditions for when the LCO should be applicable and more effectively provide appropriate exceptions to the Applicability for certain valve configurations. The amendment would incorporate other minor changes; the title to TS 3.7.3 and the header for each TS page would be revised, and the exception footnotes in TS Table 3.3.2-1 of TS 3.3.2, "ESFAS [Engineered Safety Features Actuation System] Instrumentation," would be revised to improve the application of existing notes and/or incorporate more appropriate notes as applicable.

<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 changes do not alter any design or operating limits, nor do they physically alter safety-related systems, nor do they affect the way in which safety-related systems perform their functions. The proposed changes do not change accident initiators or precursors assumed or postulated in the FSAR [Final Safety Analysis Report]-described accident analyses, nor do they alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is normally operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits. With specific regard to the proposed TS changes, although the changes involve the exceptions contained in the Applicability of TS 3.7.3 as well as the notes attached to TS Table 3.3.2-1 (which are themselves exceptions), the provisions of the exceptions and notes would continue to be based on the premise that adequate isolation or isolation capability exists for the main feedwater lines, i.e., that the required safety function is performed or capable of being performed as required or assumed for mitigation of the applicable postulated accidents.

   All accident analysis acceptance criteria will therefore continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR. The applicable radiological dose acceptance criteria will continue to be met. Overall protection system performance will remain within the bounds of the previously performed accident analyses.

   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

There are no proposed design changes, nor are there any changes in the method by which any safety-related plant structure, system, or component (SSC) performs its specified safety function. The proposed changes will not affect the normal method of plant operation or change any operating parameters. No equipment performance requirements will be affected. The proposed changes will not alter any assumptions made in the safety analyses. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment. The proposed amendment will not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

Response: No

There will be no effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The applicable radiological dose consequence acceptance criteria for design-basis transients and accidents will continue to be met. The proposed changes do not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. None of the acceptance criteria for any accident analysis will be changed.

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: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: May 4, 2009.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.7.2, "Main Steam Isolation Valves (MSIVs)," to add the main steam isolation valve bypass valves (MSIVBVs) and main steam low point drain isolation valves (MSLPDIVs) to the scope of the TS. In addition, the proposed amendment would make editorial changes to the title and header on each page of TS 3.7.2, and would incorporate other minor changes to revise exception footnote (i) in TS Table 3.3.2-1 of TS 3.3.2, "ESFAS [Engineered Safety Features Actuation System] Instrumentation," to remove the MSIVs from the footnote such that the footnote only addresses the MSIVBVs and MSLPDIVs. The MSIVs would be addressed in new exception footnote (k) added to TS Table 3.3.2-1.

The proposed amendment would add new TS 3.7.19, "Secondary System Isolation Valves (SSIVs)," which would provide limiting conditions for operation (LCOs) and surveillance requirements for the SSIVs, steam generator chemical injection isolation valves (SGCIIVs), steam generator blowdown isolation valves (SGBSIVs), and steam generator sample line isolation valves (SGBSSIVs). New Function 10, "Steam Generator Blowdown System and Sample Line Isolation Valve Actuation," would be added to TS Table 3.3.2-1. The SGBSIVs and SGBSSIVs would be addressed in new exception footnote (t) added to Table 3.3.2-1 for Function 10.

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

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

   Response: No.

   The proposed change adds requirements to the TS to ensure that systems and components are maintained consistent with the safety analysis and licensing basis.

   Requirements are incorporated into the TS for secondary system isolation valves. These changes do not involve any design or physical changes to the facility, including the SSIVs themselves. The design and functional performance requirements, operational characteristics, and reliability of the SSIVs are unchanged. There is no impact on the design safety function of MSIVs, MSIVBVs, MSLPDIVs, MFIVs [main feedwater isolation valves], MFRVs [main feedwater regulating valves] or MFRVBVs [MFRV bypass valves] to close (either as an accident mitigator or as a potential transient initiator). Since no failure mode or initiating condition that could cause an accident (including any plant transient) evaluated per the FSAR [Final Safety Analysis Report]-described safety analyses is created or affected, the change cannot involve a significant increase in the probability of an accident previously evaluated.

   With regard to the consequences of an accident and the equipment required for mitigation of the accident, the proposed changes involve no design or physical changes to components in the main steam supply system or feedwater system. There is no impact on the design safety function of MSIVs, MSIVBVs, MSLPDIVs, MFIVs, MFRVs, or MFRVBVs or any other equipment required for accident mitigation. Adequate equipment availability would continue to be required by the TS. The consequences of applicable, analyzed accidents (such as a main steam line break [or] feedline break) are not impacted by the proposed changes.

   The changes to TS 3.3.2, TS Table 3.3.2-1, and exception footnotes associated with Table Function 4 and New Function 10 maintain consistency with the Applicability of revised TS 3.7.2 and new TS 3.7.19. Maintaining TS 3.3.2 and TS Table 3.3.2-1 consistent with the Applicability of TS 3.7.2 and TS 3.7.19 is consistent with the Westinghouse Standard Technical Specifications.

   These changes involve no physical changes to the facility and do not adversely affect the availability of the safety functions assumed for the

MSIVs, MSIVBVs, MSLPDIVs, and SSIVs. Therefore, they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Based on the above considerations, 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 changes add requirements to the TS that support or ensure the availability of the safety functions assumed or required for the MSIVs, MSIVBVs, MSLPDIVs, and SSIVs. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. Additional requirements are being imposed, but they are consistent with the assumptions made in the safety analysis and licensing basis. The addition of Conditions, Required Actions and Completion Times to TS for the MSIVBVs, MSLPDIVs, and SSIVs does not involve a change in the design, configuration, or operational characteristics of the plant. Further, the proposed changes do not involve any changes in plant procedures for ensuring that the plant is operated within analyzed limits. As such, no new failure modes or mechanisms that could cause a new or different kind of accident from any previously evaluated are 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 addition of Conditions, Required Actions and Completion Times for SSIVs, MSIVBVs, and MSLPDIVs, as well as the proposed change to the LCO and Applicability for TS 3.7.2 and the proposed new TS 3.7.19 (and the corresponding changes to TS 3.3.2, "ESFAS Instrumentation") does not alter the manner in which safety limits or limiting safety system settings are determined. No changes to instrument/system actuation setpoints are involved. The safety analysis acceptance criteria are not impacted and the proposed change will not permit plant operation in a configuration outside the design basis. The changes are consistent with the safety analysis and licensing basis for the facility.

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: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: June 1, 2009.

Description of amendment request: The proposed amendment would revise the Limiting Condition for Operation (LCO) Applicability Note for Technical Specification (TS) 3.3.9, "Boron Dilution Mitigation System (BDMS)." The LCO Applicability Note would be revised to more explicitly define what the term "during reactor startup" means in MODES 2 and 3. This revision to the Applicability Note is proposed to clarify the situations during which the BDMS signal may be blocked.

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

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

   Response: No

   Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no design changes. All design, material, and construction standards that were applicable prior to this amendment request will be maintained. There will be no changes to any design or operating limits.

The proposed change will not adversely affect accident initiators or precursors [or] adversely alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. There are no design or operating changes to the reactor makeup water system (RMWS), the reactor makeup control system (RMCS), or the chemical and volume control system (CVCS). There will be no decrease in the boron concentration of the boric acid tanks. There will be no changes to the BDMS setpoint or the operation of the BDMS, other than the limited durations during which flux multiplication signal blocking would be allowed. Therefore, there will be no changes that would serve to increase the likelihood of occurrence of an inadvertent boron dilution event.

The proposed change will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the applicable acceptance limits. Exceptions to Technical Specification requirements are allowed and, in fact, rather commonplace when plant operation would otherwise be restricted in a manner that is not commensurate with the desired safety objective, especially when those exceptions are of short duration and are accompanied by compensatory measures.

The proposed change does not physically alter safety-related systems [or] affect the way in which safety-related systems perform their functions.

The inadvertent boron dilution analysis acceptance criteria will continue to be met with the proposed change, with consideration given to the fact that the current licensing basis analyses do not assume concurrent rod withdrawal in the MODES 2 and 3 boron dilution analyses. The licensing basis analyses assume that positive reactivity insertion is being added by a single method, i.e., boron dilution. The MODE 2 licensing basis analysis of an inadvertent boron dilution event in FSAR [Final Safety Analysis Report] Section 15.4.6 assumes that the shutdown banks are fully withdrawn and that the control banks are withdrawn to the 0% power rod insertion limits depicted in the COLR [Core Operating Limits Report]. The MODE 2 analysis credits operator action to swap the charging suction source after an automatic reactor trip, and corresponding rod insertion, on high source range neutron flux. The MODE 3 licensing basis analysis credits automatic mitigation by the BDMS with steady state initial conditions and static initial rod positions (all shutdown and control banks are fully inserted other than the single most reactive rod which is assumed to be fully withdrawn) at bounding RCS [reactor coolant system] T–avg values at either end of MODE 3. Neither the analysis nor the BDMS design basis assumes that the system protects against a rod withdrawal event.

The proposed change will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The applicable radiological dose criteria will continue to be met.

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

   There are [neither] proposed design changes nor are there any changes in the method by which any safety-related plant structure, system, or component (SSC) performs its specified safety function. The proposed change will not affect the normal method of plant operation or change any operating parameters. Equipment performance necessary to fulfill safety analysis missions will be unaffected. The proposed change will not alter any assumptions required to meet the safety analysis acceptance criteria.

   No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

   The proposed amendment will not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

   Response: No

   There will be no effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. Mode-specific required shutdown margins in the COLR will not be changed. The applicable radiological dose consequence acceptance criteria will continue to be met.

> The proposed change does not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. None of the acceptance criteria for any accident analysis will be changed.
>
> 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.

<u>Attorney for licensee</u>: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

<u>NRC Branch Chief</u>: Michael T. Markley.

<u>Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri</u>

<u>Date of amendment request</u>: July 10, 2009.

<u>Description of amendment request</u>: The proposed amendment would delete the Technical Specification (TS) requirements for the containment hydrogen recombiners and hydrogen monitors. The proposed TS changes support implementation of the revision to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," that became effective on October 16, 2003. The proposed changes are consistent with Revision 1 of the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-447, "Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors."

The NRC staff issued a notice of opportunity for public comments on TSTF-447, Revision 1, published in the *Federal Register* on August 2, 2002 (67 FR 50374), soliciting comments on a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination for the elimination of requirements for hydrogen recombiners, and hydrogen and oxygen monitors from TS. Based on its evaluation of the public comments received, the NRC staff made appropriate changes to the models and included final versions in a notice of availability published in the *Federal Register* on September 25, 2003 (68 FR 55416), regarding the adoption of TSTF-447, Revision 1, as part of the NRC's consolidated line item improvement process (CLIIP).

In addition to the changes related to requirements for the hydrogen recombiners and monitors, this amendment application includes four unrelated, minor changes to correct typographical errors identified in Callaway's TS.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC adopted by the licensee 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 revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.
>
> With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG [Regulatory Guide] 1.97 Category 1 is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG

1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3 and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs [severe accident management guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any 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 elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3–The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional

> probability of a large release up to approximately 24 hours after the onset of core damage.
>
> Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the [Three Mile Island], Unit 2 accident, can be adequately met without reliance on safety-related hydrogen monitors.
>
> Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

The NRC staff has reviewed the analysis adopted by the licensee 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: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Michael T. Markley.

Virginia Electric and Power Company, Docket No. 50-338 North Anna Power Station, Unit No. 1, Louisa County, Virginia

Date of amendment request: July 23, 2009

Description of amendment request: The proposed change, a one-time extension to the Completion Time (CT) of Technical Specification 3.8.9 Condition A, will provide an opportunity to fully investigate the extent of the damaged breaker and its condition to ensure continued bus reliability for the remainder of the operating cycle.

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

The proposed change does not alter any plant equipment or operating practices in such a manner that the probability of an accident is significantly increased. The proposed change will not alter assumptions relative to the mitigation of an accident or transient event. Manual operator actions in the event of a SGTR have been identified during the one-time extended CT for the 1J1 [Motor Control Center] MCC outage. A risk-informed evaluation of these operator actions has been performed and the increase in annual Core Damage and Large Early Release Frequencies associated with the proposed change in the Technical Specification CT are characterized as "small changes" by Regulatory Guide (RG) 1.174. The Incremental Conditional Core Damage and Large Early Release Probabilities [ICCDP and ICLERP] associated with the proposed Technical Specification CT meet the acceptance criteria in Regulatory Guide 1.177.

The ICCDP and ICLERP are 1.01 E-7 per year and 9.86E-9 per year, respectively. These results are below the RG 1.177 limits of 5E-7 for ICCDP and 5E-8 for ICLERP.

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. 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?

The systems' design and operation are not affected by the proposed change. The safety analysis acceptance criteria stated in the Updated Final Safety Analysis Report is not impacted by the change. Redundancy and diversity of the electrical distribution system will be maintained with the exception of the MCCs 1J 1-2N and 2S. The proposed change will not allow plant operation in a configuration outside the design basis.

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

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385

NRC Branch Chief: Undine Shoop

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.

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

Date of amendment request: March 20, 2008, as supplemented by letters dated May 28, 2008, October 6, 2008, December 17, 2008, and February 12, 2009.

Brief description of amendment request: The proposed amendments would revise the McGuire licensing basis by adopting the Alternative Source Term (AST) radiological analysis methodology as allowed by 10 CFR 50.67, Accident Source Term, for the Loss of Coolant Accident. This amendment request represents full scope implementation of the AST as described in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Revision 0."

Date of publication of individual notice in FEDERAL REGISTER: February 27, 2009 (74 FR 9009)

Expiration date of individual notice: April 28, 2009.

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

Date of amendment request: June 23, 2008.

Brief description of amendment request: The amendments revise the Technical Specifications (TSs) for Catawba Nuclear Station, Units 1 and 2. This request modifies the subject TS and Bases by changing the logic configuration of TS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation", Function 5.b. (5), "Turbine Trip and Feedwater Isolation, Feedwater Isolation, Doghouse Water Level - High High." The existing one-out-of-one (1/1) logic per train per doghouse is being modified to a two-out-of-three (2/3) logic per train per doghouse. The proposed change will improve the overall reliability of this function and will reduce the potential for spurious actuations.

Date of publication of individual notice in FEDERAL REGISTER: February 24, 2009 (74 FR 8276)

Expiration date of individual notice: April 27, 2009.

Duke Energy Carolinas, LLC, et al., Docket No. 50-414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of amendment request: November 13, 2008.

Brief description of amendment request: The amendment proposes a one-cycle revision to the Technical Specifications to incorporate an interim alternate repair criterion for steam generator tube repair criteria during the End of Cycle 16 refueling outage and subsequent cycle 17 operation.

Date of publication of individual notice in FEDERAL REGISTER: February 24, 2009 (74 FR 8278)

Expiration date of individual notice: April 27, 2009.

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: June 8, 2009.

Brief description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.9.2, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to exclude portions of the CPSES, Unit 2 Model D5 SG below the top of the SG tubesheet from periodic SG tube inspections. In addition, the proposed amendment would revise TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report," to include reporting requirements specific to the permanent alternate repair criteria for CPSES, Unit 2. The amendment request is supported by Westinghouse WCAP-17072-P, "H*: Alternate Repair Criteria for the Tube Sheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," May 2009.

Date of publication of individual notice in FEDERAL REGISTER: July 23, 2009 (74 FR 36533).

Expiration date of individual notice: September 21, 2009.

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

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

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

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

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

(ADAMS) Public Electronic Reading Room on the internet at the NRC web site, http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr@nrc.gov.

Calvert Cliffs Nuclear Power Plant, Inc., 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 23, 2009

Brief description of amendments: The amendments revise the Technical Specifications (TSs) by removing working hour restrictions from TS 5.2.2 to support compliance with recent revisions to Title 10 of the *Code of Federal Regulations*, Part 26, Subpart I. The amendments are consistent with the guidance contained in Nuclear Regulatory Commission (NRC) approved Technical Specifications Task Force Traveler 511 (TSTF-511). This TS improvement was made available by the NRC on December 30, 2008 (73 FR 79923) as part of the consolidated line item improvement process.

Date of issuance: August 6, 2009.

Effective date: As of the date of issuance to be implemented with the implementation of the new 10 CFR Part 26, Subpart I requirements.

Amendment Nos.: 292 and 268.

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 2, 2009 (74 FR 26430).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 6, 2009.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 14, 2008

Brief description of amendments: The changes revised Technical Specifications (TSs) Section 3.7.10, "Control Room Area Ventilation," its associated Bases, and TS Section 5.5 "Programs and Manuals." This LAR institutes the Control Room Habitability Program.

The changes are consistent with NRC-approved Industry Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-448, Revision 3, “Control Room Habitability Program." The availability of this TS improvement was announced in the *Federal Register* on January 17, 2007, as part of the Consolidated Line-Item Improvement Process (CLIIP). The amendments also authorized a change to the Catawba Updated Final Safety Analysis Report (UFSAR).

Date of issuance: July 30, 2009.

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

Amendment Nos.: 250 and 245.

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the licenses and the technical specifications.

Date of initial notice in FEDERAL REGISTER: June 2, 2009 (74 FR 26431).

The Commission's related evaluation, state consultation, and final no significant hazards consideration determination of the amendments is contained in a Safety Evaluation dated July 30, 2009.

No significant hazards consideration comments received: No.

Entergy Gulf States Louisiana, LLC, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 21, 2009, as supplemented by letters dated January 23 and June 22, 2009.

Brief description of amendment: The amendment modified the Technical Specifications (TSs) to adopt U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change travelers TSTF-163, TSTF-222, TSTF-230, and TSTF-306, and made two minor administrative corrections.

Date of issuance: August 11, 2009.

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

Amendment No.: 165.

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

Date of initial notice in *Federal Register*: March 24, 2009 (74 FR 12392). The supplemental letters dated January 23 and June 22, 2009, 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 amendment is contained in a Safety Evaluation dated August 11, 2009.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 1 (ANO1), Pope County, Arkansas

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2 (ANO2), Pope County, Arkansas

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1 (GGNS), Claiborne County, Mississippi

Entergy Nuclear Operations, Inc., Docket Nos. 50-247 and 50-286, Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3), Westchester County, New York

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

Entergy Nuclear Operations, Inc., Docket No. 50-293, Pilgrim Nuclear Power Station (PIL), Plymouth County, Massachusetts

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

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3 (W3), St. Charles Parish, Louisiana

Date of application for amendment: April 27, 2009, as supplemented July 10, 2009.

Brief description of amendment: The amendments deleted those portions of the Technical Specifications (TSs) superseded by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, Subpart I, consistent with U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change traveler TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26."

Date of issuance: August 4, 2009.

Effective date: As of the date of issuance and shall be implemented by October 1, 2009.

Amendment Nos.: ANO1 – 237; ANO2 – 285; JAF – 295; GGNS – 183; IP2 – 261; IP3 – 240; PAL – 238; PIL – 233; RBS – 164; and W3 – 221.

Facility Operating License Nos. DPR-51 (ANO1), NPF-6 (ANO2), DPR-59 (JAF), NPF-29 (GGNS), DPR-26 (IP2), DPR-64 (IP3), DPR-20 (PAL), DPR-35 (PIL), NPF-47 (RBS), and NPF-38 (W3): The amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: June 2, 2009 (74 FR 26432). The supplement dated July 10, 2009, 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 August 4, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 17, 2008, as supplemented by letters dated January 8, March 18, and June 30, 2009.

Brief description of amendment: The amendment revised the Operating License and modified Technical Specification (TS) 3/4.3.1 and Note 2 of TS Table 4.3-1. The changes result in the addition of conservatism to Core Protection Calculator power indications when calibrations are required in certain conditions.

Date of issuance: August 10, 2009.

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

Amendment No.: 222.

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

Date of initial notice in *Federal Register*: November 4, 2008 (73 FR 65695). The supplemental letters dated January 8, March 18, and June 30, 2009, 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 amendment is contained in a Safety Evaluation dated August 10, 2009.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

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

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

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

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

Date of application for amendments: June 9, 2008, as supplemented by letter dated March 30, 2009.

Brief description of amendments: The amendments revise the Technical Specification (TS) surveillance requirement (SR) frequency in TS 3.1.3, "Control Rod OPERABILITY." The amendments also clarify the requirement to fully insert all insertable control rods for the limiting condition for operation in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation" (Clinton Power Station only). Finally, the amendments revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

Date of issuance: August 11, 2009

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

Amendment Nos.: 188, 232/225, 193/180, 272/276, 244/239

Facility Operating License Nos. NPF- 62, DPR-19, DPR-25, NPF-11, NPF-18, DPR-44, DPR-56, DPR-29, DPR-30: The amendments revised the Technical Specifications/Licenses.

Date of initial notice in FEDERAL REGISTER: August 12, 2009 (73 FR 46928)

The March 30, 2009, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 11, 2009.

No significant hazards consideration comments received: No.

Luminant Generation Company LLC, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2 (CPSES), Somervell County, Texas

Date of amendment request: April 1, 2009, as supplemented by letter dated July 9, 2009.

Brief description of amendments: The amendments deleted Technical Specification (TS) 5.2.2.d, in TS 5.2.2, "Unit Staff," regarding the requirement to develop and implement administrative procedures to limit the working hours of personnel who perform safety-related functions. In addition, paragraphs e and f of TS 5.2.2 were renumbered to d and e and in TS 5.2.2.b the reference to 5.2.2.f was revised to 5.2.2.e to reflect the removal of paragraph d of TS 5.2.2.   The change is consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Revision 0 to TS Task Force (TSTF) Improved Technical Specification change traveler, TSTF-511, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26." The availability of this TS improvement was announced in the *Federal Register* on December 30, 2008 (73 FR 79923), as part of the consolidated line item improvement process.

Date of issuance: August 7, 2009.

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

Amendment Nos.: Unit 1 - 148; Unit 2 - 148.

Facility Operating License Nos. NPF-87 and NPF-89: The amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: May 19, 2009 (74 FR 23445). The supplemental letter dated July 9, 2009, 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 August 7, 2009.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket Nos. 50-220 and 50-410, Nine Mile Point Nuclear Station, Unit Nos. 1 and 2 (NMP 1 and 2), Oswego County, New York

Date of application for amendment: February 11, 2009.

Brief description of amendments: The amendments delete those portions of the Technical Specifications (TSs) superseded by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 26, Subpart I. This change is consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support

Compliance with 10 CFR Part 26." These changes were described in a Notice of Availability for Consolidated Line Item Improvement Process TSTF-511 published in the *Federal Register* on December 30, 2008 (73 FR 79923).

Date of issuance: July 27, 2009.

Effective date: As of the date of issuance to be implemented by October 1, 2009.

Amendment Nos.: 203 and 131.

Renewed Facility Operating License Nos. DPR-063 and NPF-069: The amendments revise the License and TSs.

Date of initial notice in FEDERAL REGISTER: April 21, 2009 (73 FR 18255).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: September 2, 2008.

Brief description of amendments: The amendment approved the licensee's request to incorporate a revision in the Updated Final Safety Analysis Report (UFSAR) Section 13.7.2.3, ''PRA Risk Categorization,'' to add a separate set of criteria for assessing the risk significance of the risk achievement worth values of common cause failures as part of the probabilistic risk assessment analysis of the risk importance of components.

Date of issuance: August 12, 2009

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

Amendment Nos.: Unit 1 - 191; Unit 2 – 179.

Facility Operating License Nos. NPF-76 and NPF-80: The amendments revised the Facility Operating Licenses, and Updated Final Safety Analysis Report.

Date of initial notice in *Federal Register*: December 2, 2008 (73 FR 73354).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 12, 2009.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendments: February 6, 2009

Brief description of amendments: The proposed amendments deleted applicable portions of the Technical Specifications (TSs) superseded by Part 26, Subpart I of Title 10 of the *Code of Federal Regulations* (10 CFR). This change is consistent with Nuclear Regulatory Commission (NRC)-approved Revision 0 to Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler, TSTF-511, "Eliminate Working Hour Restrictions from TS 5.2-2 to Support Compliance with 10 CFR Part 26."

Date of issuance: July 29, 2009.

Effective date: As of the date of issuance and shall be implemented by October 1, 2009.

Amendment Nos.: 256 and 237.

Renewed Facility Operating License Nos. NPF-4 and NPF-7: Amendments change the license and the technical specifications.

Date of initial notice in *FEDERAL REGISTER*: March 24, 2009 (74 FR 12396).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 14, 2008.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," to extend the Surveillance Frequency on selected ESFAS slave relays from 92 days to 18 months.

Date of issuance: July 30, 2009.

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

Amendment No.: 183.

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

Date of initial notice in *Federal Register*: October 7, 2008 (73 FR 58379).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 14, 2008, as supplemented by letter dated April 10, 2009.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, “Main Steam Isolation Valves (MSIVs),” and added new TS 3.7.19, “Secondary System Isolation Valves (SSIVs).” TS 3.7.2 has been revised to add MSIV bypass valves to the scope of TS 3.7.2. TS Table 3.3.2-1 has been revised to reflect the addition of the MSIV bypass valves to TS 3.7.2 and the associated applicability to be consistent with Westinghouse Standard Technical Specifications (NUREG-1431, Revision 3.0). TS 3.7.19 has been added to include a limiting condition for operation, conditions/required actions, and surveillance requirements for the steam generator blowdown isolation valves and steam generator blowdown sample isolation valves.

Date of issuance: July 31, 2009.

Effective date: As of the date of issuance and shall be implemented prior to startup from Refueling Outage 17.

Amendment No.: 184.

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

Date of initial notice in *Federal Register*: October 7, 2008 (73 FR 58679). The supplemental letter dated April 10, 2009, 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 amendment is contained in a Safety Evaluation dated July 31, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 6, 2009, as supplemented by letter dated July 14, 2009.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.2.2, "Unit Staff," to eliminate working hour restrictions (TS 5.2.2.d) to support compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26. In addition, paragraphs e and f of TS 5.2.2 were renumbered to d and e to reflect the removal of paragraph d of TS 5.2.2, and a reference in 5.2.2b was updated to reflect the renumbering of 5.2.2f. to 5.2.2e.   The request is consistent with the guidance contained in U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) change traveler TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26."

Date of issuance: August 7, 2009.

Effective date: As of its date of issuance and shall be implemented by October 1, 2009.

Amendment No.: 185.

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

Date of initial notice in *Federal Register*:   April 21, 2009 (74 FR 18258). The supplemental letter dated July 14, 2009, 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 amendment is contained in a Safety Evaluation dated August 7, 2009.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 14th day of August 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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