6. Provide sufficient information to show that a genuine dispute exists with the applicant regarding a material issue of law or fact. This information must include references to specific portions of the application (including the applicant's environmental report and safety report) that the requester/petitioner disputes and the supporting reasons for each dispute, or, if the requester/petitioner believes the application fails to contain information on a relevant matter as required by law, the identification of each failure and the supporting reasons for the requester's/petitioner's belief.

In addition, in accordance with 10 CFR 2.309(f)(2), contentions must be based on documents or other information available at the time the petition is to be filed, such as the application, supporting safety analysis report, environmental report or other supporting document filed by an applicant or licensee, or otherwise available to the petitioner. On issues arising under the National Environmental Policy Act, the requester/petitioner shall file contentions based on the applicant's environmental report. The requester/petitioner may amend those contentions or file new contentions if there are data or conclusions in the NRC draft, or final environmental impact statement, environmental assessment, or any supplements relating thereto, that differ significantly from the data or conclusions in the applicant's documents. Otherwise, contentions may be amended or new contentions filed after the initial filing only with leave of the presiding officer.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical—primarily concerns issues relating to matters discussed or referenced in the Safety Evaluation Report for the proposed action.
2. Environmental—primarily concerns issues relating to matters discussed or referenced in the Environmental Report for the proposed action.
3. Emergency Planning—primarily concerns issues relating to matters discussed or referenced in the Emergency Plan as it relates to the proposed action.
4. Physical Security—primarily concerns issues relating to matters discussed or referenced in the Physical Security Plan as it relates to the proposed action.
5. Miscellaneous—does not fall into one of the categories outlined above.

If the requester/petitioner believes a contention raises issues that cannot be classified as primarily falling into one of these categories, the requester/petitioner must set forth the contention and supporting bases, in full, separately for each category into which the requester/petitioner asserts the contention belongs, with a separate designation for that category.

Requesters/petitioners should, when possible, consult with each other in preparing contentions and combine similar subject matter concerns into a joint contention, for which one of the co-sponsoring requesters/petitioners is designated the lead representative. Further, in accordance with 10 CFR 2.309(f)(3), any requester/petitioner that wishes to adopt a contention proposed by another requester/petitioner must do so, in accordance with the E-Filing rule, within 10 days of the date the contention is filed, and designate a representative who shall have the authority to act for the requester/petitioner.

In accordance with 10 CFR 2.309(g), a request for hearing and/or petition for leave to intervene may also address the selection of the hearing procedures, taking into account the provisions of 10 CFR 2.310.

III. Further Information

Documents related to this action, including the application for amendment and supporting documentation, are available electronically at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html*. From this site, you can access the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession number for the document related to this Notice is ML073090651, Redacted Version of Amendment Request for Processing UF6 in the CD Line Facility at the NFS Site. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*.

These documents may also be viewed electronically on the public computers located at the NRC's PDR, O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee.

Dated at Rockville, Maryland, this 12th day of December 2007.

For the Nuclear Regulatory Commission.

Peter J. Habighorst,

Chief, Fuel Manufacturing Branch, Fuel Facility Licensing Directorate, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E7–25406 Filed 12–28–07; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 6, 2007 to December 19, 2007. The last biweekly notice was published on December 18, 2007 (72 FR 71703).

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

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, person(s) may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request via electronic submission through the NRC E-Filing system for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested 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 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address, and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

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

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

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for hearing or a petition for leave to intervene must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated in August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve 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 a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *HEARINGDOCKET@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 Web site 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 submitted 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 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 technical help line, which is available between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday. The help line number is (800) 397–4209 or locally, (301) 415–4737.

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, 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 petition and/or request 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). To be timely, filings must be submitted no later than 11:59 p.m. Eastern Time on the due date.

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. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

For further details with respect to this amendment action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendment request: November 19, 2007.

Description of amendment request: The proposed amendment would make administrative revisions to delete requirements that are obsolete or redundant, or correct and clarify the typing and formatting of other requirements. The proposed changes will not result in changes to the plant design or the procedural controls for the operation, surveillance, or maintenance of the plant.

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?

No. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are administrative. The changes delete obsolete or redundant requirements, clarify existing requirements, and correct typing and formatting errors. There will be no resulting changes to the plant design or procedural controls. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the Proposed Changes Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. There are no physical changes being made to the plant or to the manner in which the plant is operated. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the Proposed Changes Involve a Significant Reduction in the Margin of Safety?

No. The proposed changes do not involve a significant reduction in the margin of safety. There are no physical changes being made to the plant or to the manner in which the plant is operated. The proposed changes are administrative. The changes delete obsolete or redundant requirements, clarify existing requirements, and correct typing and formatting errors. Therefore, the changes do not involve a significant reduction in any margin of safety for HBRSEP [H.B. Robinson Steam Electric Plant], Unit No. 2.

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

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

NRC Branch Chief: Thomas H. Boyce.

Duke Power Company LLC, Docket No. 50–369, McGuire Nuclear Station, Unit 1, Mecklenburg County, North Carolina.

Date of amendment request: February 21, 2007, as supplemented August 9, 2007.

Description of amendment request: The proposed amendment would allow, on a one-time basis, an extension of the interval governing the conduct of the Integrated Leak Rate Test (ILRT) for McGuire Nuclear Station, Unit 1. The proposed amendment would revise administrative Technical Specification (TS) 5.5.2, ''Containment Leak Rate Testing Program,'' from the currently approved 15-year interval (since the last McGuire Nuclear Station, Unit 1, Type A test) to a frequency encompassing the end of the McGuire Nuclear Station, Unit 1, End-of-Cycle 19 refueling outage (approximately 6 months beyond the present TS frequency).

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

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

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed extension to the Type A testing intervals cannot increase the probability of an accident previously evaluated since extension of the intervals is not a physical plant modification that could alter the probability of accident occurrence, nor is it an activity or modification by itself that could lead to equipment failure or accident initiation. The proposed extension to the Type A testing intervals does not result in a significant increase in the consequences of an accident as documented in NUREG–1493 [''Performance-Based Containment Leak-Test Program'', NUREG–1493, September 1995]. The NUREG notes that very few potential containment leakage paths are not identified by Type B and Type C tests. It concludes that reducing the Type A testing frequency to once per twenty years leads to an imperceptible increase in risk. McGuire [Nuclear Station, Unit 1 (McGuire Unit 1)] provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. Prior Type A tests for McGuire Unit 1 identified containment leakage within acceptance criteria, indicating a very leak tight containment. Inspections required by the ASME Code [American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (Code)] are also performed in order to identify indications of containment degradation that could affect leak tightness. Separately, Type B and Type C testing, required by TS [Technical Specification] identify any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Type A test intervals will not represent a significant increase in the consequences of an accident.

Second Standard

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to the Type A test interval does not create the possibility of a new or different type of accident since there are no physical changes being made to the plants and there are no changes to the operation of the plants that could introduce a new failure mode.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to Type A test intervals will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing intervals found that a twenty-year interval resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal effect on this risk, since 95 percent of the Type A detectable leakage paths would already be detected by Type B and Type C testing. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to McGuire. Based upon the preceding discussion, Duke Energy Corporation [Duke Power Company, LLC] has concluded that the proposed amendments do not involve a significant hazards consideration.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: November 15, 2007.

Description of amendment request: The proposed change would relocate Surveillance Requirement (SR) 3.8.3.6 from the Technical Specifications (TS) to a licensee-controlled document. SR 3.8.3.6 requires the Emergency Diesel Generator (EDG) Fuel Oil Storage Tanks (FOSTs) to be drained, sediment removed, and cleaned on a 10-year interval.

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 FOSTs provide the storage for the EDG fuel oil, assuring an adequate volume is available for each EDG to operate for seven days in the event of a loss of offsite power concurrent with a loss of coolant accident. The relocation of the SR to drain and clean the FOSTs will not impact any of the previously analyzed accidents. Sediment in the tank, or failure to perform this SR, does not necessarily result in an inoperable storage tank. Fuel oil quantity and quality are assured by other TS SRs which remain unchanged. These SRs help ensure tank sediment is minimized and ensure that any degradation of the tank wall surface that results in a fuel oil volume reduction is detected and corrected in a timely manner. As a result, adequate controls exist to allow relocation of this preventative maintenance cleaning requirement to licensee controlled documents.

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 TS changes do not involve the addition or modification of any plant equipment. Also, the proposed change will not alter the design configuration, or method of operation of plant equipment beyond its normal functional capabilities. The proposed TS change does not create any new credible failure mechanisms, malfunctions or accident initiators.

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

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

Response: No.

The proposed change does not alter or exceed a design basis or safety limit. Diesel generator fuel oil quantity and quality will continue to be maintained within acceptable limits of the TS to assure the ability of the EDG to perform its intended function.

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

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

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

NRC Branch Chief: Thomas G. Hilt.

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, Claiborne County, Mississippi.

Date of amendment request: December 5, 2007.

Description of amendment request: The proposed amendment would change the Grand Gulf Nuclear Station, Unit 1 (GGNS), Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR),'' to add a reference to an analytical method that will be used to determine core operating limits. The new reference, NEDC–33383P, ''GEXL97 Correlation Applicable to ATRIUM–10 Fuel,'' will allow Entergy Operations, Inc. (Entergy) to use a Global Nuclear Fuel (GNF) method to determine fuel assembly critical power of AREVA ATRIUM–10 fuel. GGNS currently operates with a full core of ATRIUM–10 fuel. Entergy plans to use the GEXL97 correlation for GGNS operating Cycle 17 currently scheduled to begin in the fall 2008. Additionally, an administrative change is proposed to an existing reference in TS 5.6.5.

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.

Core operating limits are established each operating cycle in accordance with TS 3.2, ''Power Distribution'' and TS 5.6.5, ''Core Operating Limits Report (COLR)''. These core operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). The methods used to determine the operating limits are those previously found acceptable by the NRC and listed in TS Section 5.6.5.b.

A change to TS 5.6.5.b is requested to include an additional reference to the list of analytical methods. GGNS currently operates with a full core of AREVA ATRIUM–10 fuel but is scheduled to load GE14 fuel during the next refueling outage. GGNS plans to use the analysis methods of the new fuel vendor, GNF for the analysis of the mixed core. The GEXL97 correlation accurately models predicted core behavior and appropriately determines the overall critical power uncertainty of the method. In addition, the GEXL97 application range covers the range of expected operation of the ATRIUM–10 fuel during normal steady state and transient conditions in the GGNS reload cores. Although a depressurization transient could result in vessel pressures below the range of GEXL97, the transient would not threaten fuel cladding integrity, since the margin to the MCPR [minimum critical power ratio] safety limit increases with decreasing reactor pressure.

Additionally, Entergy proposes an administrative change to the GESTAR-Il reference in TS 5.6.5.b. The administrative change does not alter any method of analysis as described in the NRC approved versions of GESTAR-II. The requested TS changes concern the use of analytical methods and do not involve any plant modifications or operational changes that could affect any postulated accident precursors or accident mitigation systems and do not introduce any new accident initiation mechanisms. The proposed changes have no effect on the type or amount of radiation released, and have no effect on predicted offsite doses in the event of an accident. Thus, the proposed change does not affect the probability of an accident previously evaluated nor does it increase the radiological 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 kind of accident from any accident previously evaluated?

Response: No.

The proposed TS changes will not change the design function, reliability, performance, or operation of any plant systems, components, or structures. It does not create the possibility of a new failure mechanism, malfunction, or accident initiators not considered in the design and licensing bases. Plant operation will continue to be within the core operating limits that are established using NRC approved methods that are applicable to the GGNS design and the GGNS fuel.

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 adds GEXL97 to the list of analytical methods in TS 5.6.5.b that can be used to determine core operating limits. Use of the GEXL97 correlation analytical method provides an equivalent level of protection as that currently provided. The administrative change does not alter any method of analysis as described in the NRC approved versions of GESTAR-II. The proposed change does not modify the safety limits or set points at which protective actions are initiated, and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

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

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: November 16, 2007.

Description of amendment request: The proposed changes revise technical specification (TS) action requirements associated with inoperable reactor coolant system (RCS) leakage detection systems. A new TS action requirement is proposed that will address the inoperability of the drywell unit cooler condensate flow rate monitoring system concurrent with one other RCS leakage detection system, other than the primary containment atmosphere gaseous radioactivity monitoring system. This would relax the allowed out-of-service time for the specified combination of systems and is related to the current inoperability of the drywell unit cooler condensate flow rate monitoring system. The proposed changes would be effective for the remainder of the current operating cycle (Cycle 10), which is currently scheduled to end in the spring of 2009, or until the next shutdown of sufficient duration to allow for drywell unit cooler condensate flow rate monitoring system repairs, whichever comes first.

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

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

Response: No.

The proposed changes continue to maintain an acceptable level of reactor coolant system (RCS) leakage detection instrumentation required to support plant operations. The level of RCS leakage detection capability inherent with the proposed changes will continue to provide acceptable early warning detection of potential RCS pressure boundary degradation. The proposed changes do not impact the physical configuration or design function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, modified, tested, or inspected [with the exception of an increase in allowed out-of-service time for a concurrent inoperability of the drywell unit cooler condensate flow rate monitoring system and another specified RCS leakage detection system]. The proposed changes do not impact the initiators or assumptions of analyzed events, nor do they impact mitigation of accidents or transient events. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes only affect systems associated with the detection of leakage resulting from the degradation of the RCS pressure boundary. The proposed changes do not alter plant configuration or require that new plant equipment be installed. The RCS leakage detection systems will continue to function as designed in all modes of operation. No new accident type is created as a result of the proposed changes. No new failure mode for any equipment is created. The proposed changes do not alter assumptions made about accidents previously evaluated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed changes do not involve any physical changes to plant SSCs or the manner in which SSCs are operated, modified, tested, or inspected. The proposed changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions of operation, or design parameters for any SSC. The proposed changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting an accident analysis. 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, with changes as noted above, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: October 9, 2007.

Description of amendment request: A change is proposed to the technical specifications (TS) of Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, consistent with Technical Specifications Task Force (TSTF) Change Traveler TSTF–423 to the standard TSs for boiling water reactor plants, to allow, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.65(a)(4). Changes proposed herein will be made to the QCNPS, Units 1 and 2, TSs for selected required action end states providing this allowance.

The licensee reviewed the proposed no significant hazards consideration (NSHC) determination published in the **Federal Register** on March 23, 2007 (71 FR 14726) and concluded that it is applicable to QCNPS, Units 1 and 2. The licensee incorporated the proposed determination by reference to satisfy the requirements of 10 CFR 50.91(a).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC is presented below:

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

The proposed change allows a change to certain required end states when the TS Completion Times for remaining in power operation will be exceeded. Most of the requested technical specification (TS) changes are to permit an end state of hot shutdown (Mode 3) rather than an end state of cold shutdown (Mode 4) contained in the current TS. The request was limited to: (1) Those end states where entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable technical specification, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical. Risk insights from both the qualitative and quantitative risk assessments were used in specific TS assessments. Such assessments are documented in Section 6 of GE NEDC–32988, Revision 2, "Technical Justification to Support Risk Informed Modification to Selected Required Action End States for BWR Plants." They provide an integrated discussion of deterministic and probabilistic issues, focusing on specific technical specifications, which are used to support the proposed TS end state and associated restrictions. The staff finds that the risk insights support the conclusions of the specific TS assessments. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident after adopting proposed TSTF–423, are no different than the consequences of an accident prior to adopting TSTF–423. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). If risk is assessed and managed, allowing a change to certain required end states when the TS Completion Times for remaining in power operation are exceeded, i.e., entry into hot shutdown rather than cold shutdown to repair equipment, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change and the commitment by the licensee to adhere to the guidance in TSTF–IG–05–02, Implementation Guidance for TSTF–423, Revision 0, ''Technical Specifications End States, NEDC–32988–A,'' will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment, if risk is assessed and managed. The BWROG's [Boiling Water Reactor Owners Group's] risk assessment approach is comprehensive and follows staff guidance as documented in RGs [Regulatory Guides] 1.174 and 1.177. In addition, the analyses show that the criteria of the three-tiered approach for allowing TS changes are met. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A risk assessment was performed to justify the proposed TS changes. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

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

NRC Branch Chief: Russell Gibbs.

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: November 29, 2007.

Brief description of amendments: Revision to Technical Specification (TS) 3.6.7, (''Spray Additive System,'' to allow modifications to the facility potentially required to comply with U.S. Nuclear Regulatory Commission (NRC) Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accident at Pressurized Water Reactors.''

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

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

Response: No

The proposed change[s] [do] not impact the initiation or probability of occurrence of any accident.

The accidents evaluated in the Final Safety Analysis report (FSAR) that could be affected by this proposed change are those involving the pressurization of the containment and those involving recirculation of fluid within the Emergency Core Cooling System (ECCS) or the Containment Spray System (e.g., loss of coolant accidents (LOCAs)).

The change to a minimum pH [potential of Hydrogen] of 7.1 will not result in a significant increase in the radiological consequences of a LOCA as-described below.

The equilibrium spray pH during the recirculation phase resulting from this change will be greater than or equal to 7.1. The pH range for the spray will be bounded by the water spray solution which is borated water with a maximum of 2600 parts per million (ppm) boron buffered to a final spray solution pH much less than the 10.5 as described in the current FSAR Section 3.11(B) for the postulated spray solution environment. The maximum pH is the limiting parameter for equipment qualification. Since the resulting pH level will be closer to neutral using the lower limit of 7.1, post-LOCA corrosion of containment components will not be increased. Post-LOCA hydrogen generation will be reduced. There will not be an adverse radiation dose effect on any safety-related equipment. Thus, the potential for failures of the ECCS or safety-related equipment following a LOCA will not be increased as a result of the proposed change.

This modification affects the Containment Spray System which is intended to respond to and mitigate the effects of a LOCA. The Containment Spray System will continue to function in a manner consistent with the plant design basis. There will be no degradation in the performance of nor an increase in the number of challenges to equipment assumed to function during an accident situation.

Therefore, these Technical Specification (TS) revisions do not affect the probability of any event initiators. There will be no adverse changes to normal plant operating parameters, Engineered Safety Features (ESF) actuation setpoints, or accident mitigation capabilities.

The proposed change allows the Spray Additive System currently used to mitigate the consequences of an accident to maintain the equilibrium sump pH at greater than or equal to 7.1 to minimize chloride-induced stress corrosion cracking in austenitic stainless components important to safety located inside containment. Therefore, the proposed changes will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

The offsite and control room doses will continue to meet the requirements of [Title 10 of the Code of Federal Regulations (10 CFR) part 100] 10 CFR 100, 10 CFR 50 Appendix A [General Design Criterion] GDC 19, [Standard Review Plan] SRP 15.6.5.11, and SRP 6.4.11. The proposed new pH limit will provide satisfactory retention of iodine in the sump water, as well as provide adequate pH control to minimize the potential of chloride-induced stress corrosion cracking of austenitic stainless steel components.

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

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

Response: No.

The proposed change to the revised Surveillance for the Containment Spray Additive System provides for a required minimum equilibrium pH in containment post accident. There are no electrical or mechanical components being added whose failure could prevent the system from functioning.

No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of the proposed changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of this proposed change. The amount of sodium hydroxide (NaOH) will provide a minimum equilibrium sump pH of 7.1 following mixing. Therefore, the possibility of a new or different type of accident is not created.

There are no changes which would cause the malfunction of safety-related equipment, assumed to be operable in the accident analyses, as a result of the proposed Technical Specification changes. The possibility of a malfunction of safety-related equipment with a different result is not created.

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

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

Response: No

The only function of the chemical additive system is to provide pH control of the post-accident containment recirculation sump water, since the borated water from the Refueling Water Storage Tank (RWST) used as the containment spray pump suction source during injection is sufficient to remove iodine from the containment atmosphere following a LOCA. The net effect on the pH control function of reducing the amount of buffer is that the equilibrium sump pH will be lowered to a minimum of 7.1. There will be no change to the current Technical Specification acceptance limits on RWST volume and boron concentration. The resulting equilibrium sump pH level from this change will be closer to neutral; therefore, the post-LOCA corrosion of containment components will not be increased (i.e., would be reduced).

Because the long term pH will be maintained greater than or equal to 7.1, margin to minimize the potential for stress corrosion cracking is maintained.

The radiological analysis, as discussed in the technical analysis above, is shown not to be impacted. There will be no change to the [departure from nucleate boiling ratio] DNBR Correlation Limit, the design DNBR limits, or the safety analysis DNBR limits discussed in Bases Section 2.1.1. There will be no effect on the manner in which Safety Limits or Limiting Safety System Settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no adverse impact on Departure of Nucleate Boiling Ratio limits, [heat flux hot channel factor] F_Q, [nuclear enthalpy rise hot channel factor] F-delta-H, LOCA peak cladding temperature, peak local power density, or any other margin of safety.

Therefore the proposed change[s] [do] not involve a reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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 requests: October 2, 2007.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3.5.4, ''Refueling Water Storage Tank (RWST),'' Surveillance Requirement (SR) 3.5.4.2, to increase the minimum required borated water volume from ''≥ [greater than or equal to] 400,000 gallons (81.5% indicated level)'' to ''≥ 455,300 gallons (93.6% level),'' to reflect the new sump design required to comply with U.S. Nuclear Regulatory Commission (NRC) Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accident at Pressurized Water Reactors.''

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

1. [Do] the proposed change[s] involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change[s] [revise] the minimum RWST borated water volume. The RWST borated water volume is not an initiator of any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected. The proposed change[s] [do] not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The effect on containment flood level, equipment qualification, and containment sump pH remain within the limits assumed in the design and accident analyses. The proposed change[s] [do] not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change[s] [do] not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change[s] are consistent with the safety analysis assumptions and resultant consequences.

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

2. [Do] the proposed change[s] create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The change[s] [do] not involve a physical alteration of the plant (i.e., no new or different components or physical changes are involved with this change) or a change in the methods governing normal plant operation. The change[s] [do] not alter any assumptions made in the safety analysis.

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

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

Response: No.

The proposed change[s] to revise the required RWST minimum borated water volume [do] not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by [these] change[s]. The proposed change[s] will not result in plant operation in a configuration outside of the design basis.

Therefore, the proposed change[s] [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 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: Thomas G. Hiltz.

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

Date of amendment request: November 5, 2007.

Description of amendment request: The licensee has proposed amending the technical specifications (TS) to delete many operational and administrative requirements upon transfer of spent nuclear fuel assemblies and fuel fragment containers from the Spent Fuel Pool (SFP) to the Humboldt Bay Independent Spent Fuel Storage Installation (ISFSI). Some TS requirements will be relocated to the HBPP Quality Assurance Plan.

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

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

Response: No.

The proposed changes reflect the transfer of spent fuel from the Spent Fuel Pool to the Humboldt Bay (HB) Independent Spent Fuel Storage Installation. Design basis accidents related to the SFP are discussed in the Humboldt Bay Power Plant Unit 3 Defueled Safety Analysis Report (DSAR). These postulated accidents are predicated on spent fuel being stored in the SFP. With the removal of the spent fuel from the SFP, there are no important-to-safety systems, structures or components required to function or to be monitored. In addition, there are no remaining credible accidents involving spent fuel or the SFP that require actions of a Certified Fuel Handler or Noncertified Fuel Handler to prevent occurrence or to mitigate consequences. The proposed change to the Design Features section of the Technical Specifications (TS) clarifies that the spent fuel is being stored in dry casks within an ISFSI. The probability or consequences of accidents at the ISFSI are evaluated in the HB ISFSI Final Safety Analysis Report (FSAR) and are independent of the accidents evaluated in the HBPP Unit 3 DSAR. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes reflect the reduced operational risks as a result of the spent fuel being transferred to dry casks within an ISFSI. The proposed changes do not modify any systems, structures or components. The plant conditions for which the HBPP Unit 3 DSAR design basis accidents relating to spent fuel and the SFP have been evaluated are no longer applicable. The aforementioned proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of an accident. Design basis accidents associated with the dry cask storage of spent fuel are already considered in the HB ISFSI FSAR. No new accident scenarios are created as a result of deleting nonapplicable operational and administrative requirements. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from those previously evaluated.

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

Response: No.

The proposed changes reflect the reduced operational risks as a result of the spent fuel being transferred to dry casks within an ISFSI. The design basis and accident assumptions within the HBPP Unit 3 DSAR and the TS relating to spent fuel are no longer applicable. The proposed changes do not affect remaining plant operations, nor structures, systems, or components supporting decommissioning activities. In addition, the proposed changes do not result in a change in initial conditions, system response time, or in any other parameter affecting the course of a decommissioning activity accident analysis. Therefore, the proposed changes will not involve a significant reduction in the margin of safety.

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

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

NRC Branch Chief: Andrew Persinko.

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

Date of amendment request: October 31, 2007.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.8.1, ''Essential Service Water System (ESW),'' and TS 3.8.1, ''AC [Alternating Current] Sources—Operating.'' A note would be added to Condition A, one ESW train inoperable, of TS 3.8.1, and Condition B, one diesel generator (DG) inoperable, of TS 3.8.1 would be revised. The revisions are to allow a one-time completion time extension from 72 hours to 14 days to support a planned replacement of ESW piping prior to December 31, 2008, in the licensee's fall 2008 refueling outage.

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 only change to the plant is that existing ESW piping will be replaced in the fall 2008 refueling outage. There are no other changes to the plant and no hardware or equipment will be added to the plant. This replacement is to address localized degradation of the ESW piping due to microbiologically induced corrosion.]

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed to the protection systems. The same reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. The use of polyethylene (PE) piping [(i.e., replacing existing ESW piping by PE piping)] in the ESW system in accordance with ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] Code Case N–755, with justified materials and design exceptions as noted in [the licensee's letter dated August 30, 2007 (ULNRC–05434), which requested relief from the ASME Code to replace the ESW piping by the PE piping], will [have the PE piping that replaces the ESW piping] provide an acceptable level of quality and safety. There will be no changes to the essential service water (ESW) system or [the] ultimate heat sink (UHS) surveillance and operating limits. [The licensee's letter dated August 30, 2007,] demonstrates the acceptability of using the PE piping in this buried ASME Class 3 application [(i.e., replacing existing ESW piping)].

The proposed changes will not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configurations of the facility or the manner in which the plant is operated and maintained. The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended [safety] functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed changes do not affect the way in which safety-related systems perform their [safety] functions.

All accident analysis acceptance criteria will 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 [Final Safety Analysis Report for the Callaway Plant].

The applicable radiological dose acceptance criteria [is unchanged] and will continue to be met.

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 changes in the method by which any safety-related plant SSC performs its safety function. [The proposed changes will not affect the performance of the ESW piping in terms of providing mitigation of design basis accidents per the FSAR accident analyses.] 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 [kind of] 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 will continue to be met. [The proposed changes will not affect the performance of the ESW piping in terms of providing mitigation of design basis accidents per the FSAR accident analyses.]

The proposed changes do not eliminate any surveillances or alter the frequency of [any] surveillances required by the Technical Specifications. None of the acceptance criteria for any accident analyses will be changed.

The proposed changes will have no impact on the radiological consequences of a design basis accident.

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, NW., Washington, DC 20037.

NRC Branch Chief: Thomas G. Hiltz.

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.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida.

Date of application for amendment: November 12, 2007.

Brief description of amendment: Use of alternate method of monitoring rod position for a control rod or shutdown rod with an inoperable rod position indicator.

Date of publication of individual notice in the ***Federal Register***: November 28, 2007 (72 FR 67323).

Expiration date of individual notice: December 28, 2007 (Public comments) and January 28, 2008 (Hearing requests).

Notice of Issuance of amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application of amendments: January 31, 2007.

Brief description of amendments: The amendments revised the Technical Specifications to remove requirements that are no longer applicable due to the completion of the control room intake/booster fan modifications.

Date of Issuance: December 11, 2007.

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

Amendment Nos.: 358, 360, and 359.

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

Date of initial notice in ***Federal Register***: October 9, 2007 (72 FR 57353) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 11, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 30, 2007, as supplemented by letter dated November 6, 2007.

Brief description of amendment: The changes revise Technical Specification (TS) 1.4, ''Frequency,'' TS 3.1.5, ''Control Rod Scram Accumulators,'' TS 3.4.1, ''Recirculation Loops Operating,'' TS 3.5.1, ''ECCS [Emergency Core Cooling System]—Operating,'' TS 3.5.2, ''ECCS—Shutdown,'' TS 3.7.1, ''Standby Service Water (SW) System and Ultimate Heat Sink (UHS),'' TS 3.8.1, ''AC [Alternating Current] Sources—Operating,'' TS 3.8.2, ''AC Sources—Shutdown,'' and TS 5.5.6, ''In-service Testing Program.'' The changes include updates to adopt approved TS Task Force (TSTF) Travelers 284, Revision 3, ''Add 'Met' vs. 'Perform' to Specification 1.4, Frequency,'' TSTF–479, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a,'' and TSTF–485, Revision 0, ''Correct Example 1.4–1,'' and TSTF–497, Revision 0, ''Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less.''

Date of issuance: December 13, 2007.

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

Amendment No.: 205.

Facility Operating License No. NPF–21: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register***: August 28, 2007 (72 FR 49572). The supplement dated November 6, 2007, 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 initially published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 13, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 2, 2007.

Brief description of amendment: The amendment modified River Bend Station, Unit 1, technical specifications (TSs) requirements for MODE change limitations in Limiting Condition for Operation 3.0.4 and Surveillance Requirement 3.0.4. The TS changes are consistent with Revision 9 of NRC-approved Industry TS Task Force (TSTF) Standard TS Change Traveler, TSTF–359, ''Increase Flexibility in MODE Restraints.'' In addition, the amendment also changed TS Section 1.4, ''Frequency,'' Example 1.4–1, ''Surveillance Requirements,'' to accurately reflect the changes made by TSTF–359, which is consistent with NRC-approved TSTF–485, Revision 0, ''Correct Example 1.4–1.''

Date of issuance: December 6, 2007.

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

Amendment No.: 156.

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

Date of initial notice in ***Federal Register***: September 11, 2007 (72 FR 51856).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 6, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 15, 2006, as supplemented on July 25 and October 9, 2007.

Brief description of amendments: The amendments revised the DCCNP–1 and DCCNP–2 Technical Specifications (TS) to allow certain functions in the reactor protection system and engineered safety feature actuation system instrumentation which have installed bypass test capability to be tested in bypass. The licensee's request to correct the administrative error will be reviewed and resolved by separate correspondence.

Date of issuance: December 17, 2007.

Effective date: As of the date of issuance, and shall be implemented within 45 days.

Amendment No.: 300, 283.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revise the License Page and Technical Specifications.

Date of initial notice in ***Federal Register***: November 21, 2006 (71 FR 67396)

The supplemental letters contained clarifying information, did not change the initial no significant hazards consideration determination, and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a safety evaluation dated December 17, 2007.

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, Somervell County, Texas.

Date of amendment request: December 19, 2006.

Brief description of amendments: The amendments revised Technical Specification 5.5.16, ''Containment Leakage Rate Testing Program,'' for consistency with the requirements of paragraph 50.55a(g)(4) of Title 10 of the Code of Federal Regulations for components classified as Code Class CC.

Date of issuance: December 13, 2007.

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

Amendment Nos.: Unit 1–141; Unit 2–141.

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 8, 2007 (72 FR 26179). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 16, 2007, as supplemented by letter date November 5, 2007.

Brief description of amendment: The amendment revised Technical Specification 5.5.6, ''Inservice Testing Program,'' to allow a one-time extension of the 5-year frequency requirement for setpoint testing of safety valve MS–RV–70ARV.

Date of issuance: December 4, 2007.

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

Amendment No.: 228.

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

Date of initial notice in ***Federal Register***: September 25, 2007 (72 FR 54476). The supplement dated November 5, 2007, 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 initially published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 4, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 28, 2007, as supplemented by letter dated May 22, 2007.

Brief description of amendments: The amendments revise the language in the Technical Specifications to conform to the licensing basis as established by Amendment Nos. 87 and 74, for Units 1 and 2, respectively, dated May 27, 1997.

Date of issuance: December 6, 2007.

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

Amendment Nos.: Unit 1–181; Unit 2–168.

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

Date of initial notice in ***Federal Register***: May 22, 2007 (72 FR 28723). The supplement dated May 22, 2007, 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 December 6, 2007.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, Goodhue County, Minnesota.

Date of application for amendments: December 14, 2006, supplemented by letter dated November 13, 2007.

Brief description of amendments: The amendments revise the sump debris interceptor nomenclature in PINGP Unit 1 and Unit 2 Technical Specifications (TS) 3.5.2 to more clearly reflect the configuration of the new Emergency Core Cooling System sump strainers that were installed to address Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors.'' The amendments also revise the required Refueling Water Storage Tank (RWST) water level in TS 3.5.4 to reflect the administratively controlled water inventory in the RWST.

Date of issuance: December 14, 2007.

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

Amendment Nos.: 182/172.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Facility Operations License and Technical Specifications.

*Date of initial notice in **Federal Register**:* February 27, 2007 (72 FR 8804)

The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 14, 2007.

No significant hazards consideration comments received: No.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska.

Date of amendment request: July 31, 2007.

Brief description of amendment: The amendment revises Technical Specification (TS) 2.7(1), (Electrical Systems—Minimum Requirements,'' TS 2.7(2), (''Electrical Systems—Modification of Minimum Requirements,'' and TS 3.7(5), ''Emergency Power System Periodic Tests—Required Safety Related Inverters.'' The licensee is adding two safety-related swing inverters to the 120 Volt alternating current instrument buses. The TS changes reflect modifications made to the plant and are needed to take advantage of the additional operational flexibility the swing inverters will provide.

Date of issuance: December 17, 2007.

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

Amendment No.: 251.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* August 28, 2007 (72 FR 49582). The Commission's related evaluation of the amendment is contained in a safety evaluation dated December 17, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 21st day of December, 2007.

For The Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–25416 Filed 12–28–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Security Officer Attentiveness

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of issuance.

SUMMARY: All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operation and have certified that fuel has been removed from the reactor vessel, and Category I fuel facilities. The contents of this bulletin are for information to Category III fuel facilities, independent spent fuel storage installations, conversion facilities and gaseous diffusion plants. The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to achieve the following three objectives:

1. The agency is notifying addressees about the NRC staff's need for information associated with licensee security program administrative and management controls as a result of security personnel inattentiveness, especially involving complicity, and related concerns with the behavior observation program (BOP). The information is needed to determine if further regulatory action is warranted, if the necessary inspection program needs to be enhanced, or if additional assessment of security program implementation is needed.

2. The NRC seeks to obtain information on licensee administrative and managerial controls to deter and address inattentiveness and complicity among licensee security personnel including contractors and subcontractors.

3. This bulletin requires that addressees provide a written response to the NRC in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) or 10 CFR 70.22(d).

This **Federal Register** notice is available through the NRC's Agencywide Documents Access and Management System (ADAMS) under accession number ML073480342.

DATES: The bulletin was issued on December 12, 2007.

ADDRESSES: Not applicable.

FOR FURTHER INFORMATION CONTACT: Timothy S. McCune at 301–415–6474 or by email *tsm5@nrc.gov*, Kevin Ramsey at 301–415–3123 or by e-mail *kmr@nrc.gov*, or Merrilee Banic at 301–415–2771 or email *mjb@nrc.gov*.

SUPPLEMENTARY INFORMATION: NRC Bulletin 2007–01 may be examined, and/or copied for a fee, at the NRC's Public Document Room at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html*. The ADAMS number for the bulletin is ML051740058.

If you do not have access to ADAMS or if you have problems in accessing the documents in ADAMS, contact the NRC Public Document Room (PDR) reference staff at 1–800–397–4209 or 301–415–4737 or by e-mail to *pdr@nrc.gov*.

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

For the Nuclear Regulatory Commission.

Martin C. Murphy,

Chief, Generic Communications Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation.

[FR Doc. E7–25398 Filed 12–28–07; 8:45 am]

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