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 February 28, 2008 to March 12, 2008. The last biweekly notice was published on March 11, 2008 (73 FR 13021).

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.

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$_{TM}$ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer$_{TM}$ 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 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 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.*

Duke Power Company 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: December 11, 2007.

Description of amendment request: The amendments would revise the

Technical Specifications (TSs) permitting relaxation of the allowed bypass test times and completion times for various systems in accordance with Technical Specification Task Force Traveler (TSTF) 418, Revision 2, ''RPS and ESFAS Test Times and Completion Times (WCAP–14333).

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:

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

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

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

The determination on risk impacts that the results of the proposed changes are acceptable was established in the NRC Safety Evaluations prepared for WCAP–14333–P–A (issued by letter dated July 15, 1998) and for WCAP–15376–P–A (issued by letter dated December 20, 2002). Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions.

The proposed changes based on TSTF–246 do not involve any physical alteration of plant SSCs. The remaining intermediate range and power range nuclear instruments remain operable and have required actions that ensure compliance with applicable safety analyses.

Therefore, it is concluded that this change does not increase the probability of occurrence of a malfunction of equipment important to safety.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not result in a change in the manner in which the RTS or ESFAS provide plant protection. The RTS and ESFAS will continue to have the same setpoints after the proposed changes are implemented. There are no design changes associated with the license amendment. The changes to Completion Times, bypass test times, and Surveillance Frequencies do not change any existing accident scenarios, nor create any new or different accident scenarios. The changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

The proposed changes do not introduce new failure mechanisms for systems, structures, or components not already considered in the UFSAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created because no new failure mechanisms or initiating events have been introduced.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and ESFAS is also maintained. Signals credited as primary or secondary and operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Times and bypass test time, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

a. Reduced testing will result in fewer inadvertent reactor trips, less frequent actuation of ESFAS components, less frequent distraction of operations personnel without significantly affecting RTS and ESFAS reliability.

b. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to instrumentation Required Actions with short Completion Times.

c. Longer repair times associated with increased Completion Times will lead to higher quality repairs and improved reliability.

d. The Completion Time extensions for the reactor trip breakers will provide the utilities additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker Completion Times, and provide consistency with the Completion Times for the logic trains.

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

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

Attorney for licensee: 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: Melanie Wong.

Duke Power Company LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: December 11, 2007.

Description of amendment request: The proposed amendments would revise the Technical Specifications permitting relaxation of the allowed bypass test times and completion times for various systems in accordance with Technical Specification Task Force Traveler (TSTF) 418, Revision 2, ''RPS and ESFAS Test Times and Completion Times (WCAP–14333).

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:

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

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

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

The determination that the results of the proposed changes are acceptable was established in the NRC Safety Evaluations prepared for WCAP–14333–P–A (issued by letter dated July 15, 1998) and for WCAP–15376–P–A (issued by letter dated December 20, 2002). Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions.

The proposed changes based on TSTF–246 do not involve any physical alteration of plant systems, structures, or components. The remaining intermediate range and power range nuclear instruments remain operable and have required actions that ensure compliance with applicable safety analyses.

Therefore, it is concluded that this change does not increase the probability of occurrence of a malfunction of equipment important to safety.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not result in a change in the manner in which the RTS or ESFAS provide plant protection. The RTS and ESFAS will continue to have the same setpoints after the proposed changes are implemented. There are no design changes associated with the license amendment. The changes to Completion Times, bypass test times, and Surveillance Frequencies do not change any existing accident scenarios, nor create any new or different accident scenarios. The changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

The proposed changes do not introduce new failure mechanisms for systems, structures, or components not already considered in the UFSAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created because no new failure mechanisms or initiating events have been introduced.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and ESFAS is also maintained. Signals credited as primary or secondary and operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Times and bypass test time, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

e. Reduced testing will result in fewer inadvertent reactor trips, less frequent actuation of ESFAS components, less frequent distraction of operations personnel without significantly affecting RTS and ESFAS reliability.

f. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to instrumentation Required Actions with short Completion Times.

g. Longer repair times associated with increased Completion Times will lead to higher quality repairs and improved reliability.

h. The Completion Time extensions for the reactor trip breakers will provide the utilities additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker Completion Times, and provide consistency with the Completion Times for the logic trains.

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

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

Attorney for licensee: 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: Melanie Wong.

Duke Power Company LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 22, 2008.

Description of amendment request: The proposed amendments would modify Technical Specification (TS) requirements related to control room envelope habitability in accordance with Technical Specification Task Force (TSTF)–448, Revision 3, ''Control Room Habitability.'' For McGuire Nuclear Station, Units 1 and 2, this TSTF revises TS 3.7.9, Control Room Area Ventilation System (CRAVS), and adds a new administrative controls program, TS 5.5.16, Control Room Envelope Habitability Program.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075) on possible license amendments adopting TSTF–448 using the NRC's consolidated line item improvement process (CLIIP) for amending the licensee's TSs, which included a model safety evaluation (SE) and model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on January 17, 2007 (72 FR 2022), which included the resolution of public comments on the model SE. The licensee has affirmed the applicability of the following NSHC determination in its application.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the control room envelope (CRE) emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Accident Previously Evaluated

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

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

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: Melanie C. Wong.

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2 (IP2), Westchester County, New York

Date of amendment request: December 13, 2007.

Description of amendment request: The proposed amendment would add some Emergency Core Cooling System (ECCS) valves and remove other ECCS valves from Surveillance Requirement (SR) 3.5.2.1. The purpose of the SR is to verify that ECCS valves whose single failure could cause loss of the ECCS function are in the required position with power removed so that the single failure could not occur. The valves being added are currently controlled administratively. The valves being removed have been evaluated to demonstrate that a single failure would not cause loss of the ECCS function.

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 adds three ECCS valves and removes four ECCS valves from IP2 SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with power removed so that misalignment or single failure cannot prevent completion of the ECCS function. The performance of the SR does not involve any actions related to the initiation of an accident and therefore the proposed changes cannot increase the probability of an accident. Misalignment or single failure of one of the three valves being added to TS could cause a loss of the ECCS function so the change will not increase the consequences of an accident but rather provide assurance that no such increase can occur. Removal of the four valves has been evaluated and the evaluation demonstrates that the misalignment or single failure of one of the valves will not affect the ECCS function and therefore will not increase the consequences of 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 change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change adds three ECCS valves and removes four ECCS valves from IP2 SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with power removed so that misalignment or single failure cannot prevent completion of the ECCS function. The removal of valves from the surveillance allows power to be maintained to the valves during normal operation but does not otherwise affect the function of the valves or the design and operation of plant systems. The addition of power does mean that the valves could fail open but this does not create the possibility of a new or different type of accident since such a failure mode is currently evaluated. The performance of the SR for added valves does not affect the function of the valves or the manner in which the valves or their systems are operated or any procedures used for valve or system operation. The change assures that the valves will be in their correct position and does not introduce any new failure modes or the possibility of a different accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change adds three ECCS valves and removes four ECCS valves from IP2 SR 3.5.2.1. The purpose of the surveillance is to assure that the valves are in their required position with power removed so that misalignment or single failure cannot prevent completion of the ECCS function. The addition of the three valves to the TS provides additional assurance that operation will be with power removed and the valves in the correct position. This increases safety margin. Removal of valves from the surveillance is based on analysis of the effects of misalignment or single failure on the ECCS function. Analysis demonstrates that the misalignment or single failure would not adversely affect the ECCS function and therefore there is no significant reduction in the margin of safety. The margin of safety remains adequate to assure the ECCS function is performed.

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

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

NRC Branch Chief: Mark G. Kowal.

Entergy Nuclear Operations, Inc., Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3, Westchester County, New York

Date of amendment request: December 18, 2007.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) requirements related to control room envelope habitability by adding a Control Room Envelope Habitability Program and then referencing this program in place of existing surveillances. It also standardizes terminology and modifies other TS related to the control room envelope.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–448, Revision 3. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075), on possible amendments concerning TSTF–448, including a model safety evaluation and model no significant hazards (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on January 17, 2007 (72 FR 2022). The licensee affirmed the applicability of the following NSHC determination in its application dated December 18, 2007.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased.

Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice.

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

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

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

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

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

NRC Branch Chief: Mark G. Kowal.

Entergy Nuclear Operations, Inc., Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3, Westchester County, New York

Date of amendment request: December 20, 2007.

Description of amendment request: The proposed amendment would modify Technical Specifications (TS), to replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on DOSE EQUIVALENT XE–133 and would take into account only the noble gas activity in the primary coolant.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–490. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 20, 2006 (71 FR 67170), on possible amendments concerning TSTF–490, including a model safety evaluation and model no significant hazards (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 15, 2007 (72 FR 12217). The licensee affirmed the applicability of the following NSHC determination in its application dated December 20, 2007.

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

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

Reactor coolant specific activity is not an initiator for any accident previously evaluated. The Completion Time when primary coolant gross activity is not within limit is not an initiator for any accident previously evaluated. The current variable limit on primary coolant iodine concentration is not an initiator to any accident previously evaluated. As a result, the proposed change does not significantly increase the probability of an accident. The proposed change will limit primary coolant noble gases to concentrations consistent with the accident analyses. The proposed change to the Completion Time has no impact on the consequences of any design basis accident since the consequences of an accident during the extended Completion Time are the same as the consequences of an accident during the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

The proposed change in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for 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 proposed change revises the limits on noble gas radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses.

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

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: January 31, 2008.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to change the description of fuel assemblies specified in TS 4.2.1, and add the Framatome Advanced Nuclear Power, Inc. (ANP) report, BAW–10240(P)–A, ''Incorporation of M5 Properties in Framatome ANP Approved Methods,'' to the analytical methods referenced in TS 5.6.5.b to permit the use of M5 alloy for fuel rod cladding and fuel assembly structural components in future operating cycles.

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 license amendment adds a Nuclear Regulatory Commission approved analytical method, BAW–10240(P)–A, ''Incorporation of M5 Properties in Framatome ANP Approved Methods,'' used to determine the core operating limits, to Technical Specification (TS) 5.6.5.b and changes the description of fuel assemblies specified in TS 4.2.1 to allow use of the M5 alloy. The proposed amendment does not affect the acceptance criteria for any Final Safety Analysis Report (FSAR) safety analysis analyzed accidents and anticipated operational occurrences. As such, the proposed amendment does not increase the probability or consequences of an accident. The proposed amendment does not involve operation of the required structures, systems or components (SSCs) in a manner or configuration different from those previously recognized or evaluated.

Therefore, operation of the facility in accordance with the proposed amendment would 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.

Use of M5 clad fuel will not result in changes in the operation or configuration of the facility. Topical report BAW–10240(P)–A describes, by reference, that the material properties of the M5 alloy are similar or better than those of zircaloy-4. Therefore, M5 fuel rod cladding and fuel assembly structural components will perform similarly to those fabricated from zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

Since the material properties of M5 alloy are similar or better than those of zircaloy-4, there will be no significant changes in the types of any effluents that may be released off-site. There will not be a significant increase in occupational or public radiation exposure.

The proposed amendment does not involve operation of any required SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the changes being requested.

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

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

Response: No.

The proposed change will not involve a significant reduction in the margin of safety because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of zircaloy-4. M5 alloy is expected to perform similarly or better than zircaloy-4 for all normal operating and accident scenarios, including both loss-of-coolant accident (LOCA) and non-LOCA scenarios. The proposed changes do not affect the acceptance criteria for any FSAR safety analysis analyzed accidents or anticipated operational occurrences. All required safety limits would continue to be analyzed using methodologies approved by the Nuclear Regulatory Commission.

Therefore, the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Acting Branch Chief: Patrick D. Milano.

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: February 1, 2008.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 5.5.16.a, ''Containment Leakage Rate Testing Program,'' to add an exception to Regulatory Guide 1.163 to allow the use of Standard ANSI/ANS 56.8–2002, and to revise TS 5.5.16.b to specify both a lower peak calculated containment internal pressure following a large-break loss-of-coolant accident (LOCA) and containment design pressure.

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 to TS 5.5.16.a adds an exception to Regulatory Guide 1.163 to specify use of Standard ANSI/ANS–56.8–2002, rather than ANSI/ANS–56.8–1994.

The proposed change to TS 5.5.16.b specifies both the peak calculated containment internal pressure with margin following a large-break LOCA and the containment design pressure.

These changes only affect the applicable version of the standard (2002 in place of 1994) and the test pressures for containment leak-rate tests, and do not involve the modification of any plant equipment or have any effect on plant operation. The changes are made based on the safety analysis and containment design, and do not have any adverse effect on accidents previously evaluated.

Therefore, the proposed changes 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 accident from any accident previously evaluated?

Response: No.

The proposed changes do not involve a physical alteration to the plant or a change in the methods governing normal plant operation. The changes are made based on the safety analysis and containment design, and do not affect any previously evaluated accidents.

Therefore, the proposed change[s] [do] not create the possibility of a new or different 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 changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes, and the changes will not result in plant operation in a configuration outside the design basis.

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

NRC Branch Chief: Thomas G. Hiltz.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: February 29, 2008.

Description of amendment request: The proposed amendments would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in accordance with the Nuclear Regulatory Commission (NRC)-approved Revision 3 of Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF–448, ''Control Room Habitability.''

The NRC staff published a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075), on possible license amendments adopting TSTF–448 using the NRC's consolidated line-item improvement process (CLIIP) for amending licensees' TSs, which included a model safety evaluation (SE) and model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on January 17, 2007 (72 FR 2022), which included the resolution of public comments on the model SE and model NSHC determination. The licensee affirmed the applicability of the following NSHC determination in its application dated February 29, 2008.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Accident Previously Evaluated

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation as determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Melanie C. Wong.

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: January 28, 2008.

Description of amendment request: The amendments would revise the Technical Specifications (TS) to establish an Action in TS 3.3.1, ''Reactor Trip Instrumentation,'' for two inoperable channels of extended range neutron flux instrumentation. The licensee also proposes a minor correction to revise ACTION c of TS 3.4.1.4.2, ''Reactor Coolant System, Cold Shutdown—Loops Not Filled,'' to change the requirement for verification of boron concentration to verification of shutdown margin.

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 extended range neutron flux monitoring instrumentation that is the subject of the proposed change performs a monitoring function and of itself has no potential as an accident initiator. The proposed requirement for the condition where both channels of the function are inoperable establishes actions that preserve the design basis where no actions previously existed. This is a more restrictive change and thus does not increase the probability or consequences of an accident previously evaluated.

The proposed change[s] to TS 3.4.1.4.2 ACTION c. clarification regarding the verification of shutdown margin [do] not result in any technical change in the way the TS ACTION is applied. Therefore this proposed change does not increase the probability or consequences of an accident previously evaluated.

The proposed change[s] [include] formatting changes that are administrative and consequently have no effect on accident analyses.

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 proposed changes do not involve any physical alteration of plant equipment and [do] not change the method by which any safety related structure, system, or component performs its function or is tested. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

The proposed change[s] [include] formatting changes that are administrative and consequently have no effect on accident analyses.

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 changes do not negate any existing requirement, and d[o] not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The purpose of the proposed changes is to provide greater assurance that the design basis is maintained. There are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change[s].

The proposed change[s] [include] formatting changes that are administrative and consequently have no effect on accident analyses.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: January 23, 2008.

Description of amendment request: The amendments would revise the Technical Specification (TS) 3.6.1.3 Actions to (1) allow entry and exit through the containment air lock doors, even if the applicable action requires the containment air lock door to be closed, and (2) expand the current guidance provided to address inoperable air lock components.

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 Technical Specification changes to revise the action requirements associated with the containment air lock will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The containment air lock is not an accident initiator. The proposed changes will not revise the operability requirements (e.g., leakage limits) for the containment air lock. Proper operation of the containment air lock will still be verified. As a result, the design basis accidents will remain the same postulated events described in the South Texas Project Unit 1 and Unit 2 Updated Final Safety Analysis Report, and the consequences of the design basis accidents will remain the same.

Therefore, the proposed changes will not increase 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 proposed changes to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed changes do not introduce any new failure modes.

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

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

Response: No.

The proposed Technical Specification changes to revise the action requirements associated with the containment air lock will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The operability requirements for the containment air lock have not been changed. The containment air lock will continue to function as assumed in the safety analysis. In addition, the proposed changes will not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety.

Therefore, the proposed changes will not result in 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

NRC Branch Chief: Thomas G. Hiltz

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

Date of amendment request: December 28, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification Administrative Controls Section 5.5.8, ''Inservice Testing Program,'' to indicate that the Inservice Testing Program shall include testing frequencies applicable to the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance, and to indicate that there may be some non-standard frequencies specified as 2 years or less in the Inservice Testing Program to which the provisions of Surveillance Requirement 3.0.2 is applicable.

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 revises TS 5.5.8, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves. 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 change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events, nor does it involve the addition or removal of any equipment, or any design changes to the facility. Therefore, the proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change revises TS 5.5.8, ''lnservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves. 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 change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site, and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No

The proposed change revises TS 5.5.8, ''Inservice Testing Program, '' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety functions of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: December 28, 2007.

Description of amendment request: The amendment would revise Technical Specifications (TS) 3.7.2, to add the Main steam isolation valve (MSIV) bypass valves to the scope of the TS. The proposed changes include a revision to the APPLICABILITY for the TS and a revision to footnote (i) in Table 3.3.2–1 of TS 3.3.2, ''ESFAS Instrumentation,'' to make it consistent with the revised Applicability of LCO 3.7.2. The amendment would also add new TS 3.7.19, ''Secondary System Isolation Valves (SSIVs),'' to include Limiting Conditions for Operation and Surveillance Requirements for the secondary system isolation valves: Main steam low point drain isolation valves, steam generator chemical injection isolation valves, steam generator blowdown isolation valves, and steam generator sample line isolation valves.

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, MFIVs, MFRVs or MFRVBVs [main steam isolation valves, main feedwater isolation valves, main feedwater regulating valves, main isolation feedwater regulating valve 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, 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 of feedline break) are not impacted by the proposed changes.

The change in APPLICABILITY for the MSIVs is consistent with the Westinghouse Standard Technical Specification 3.7.2. The change to footnote (i) in TS Table 3.3.2–1 makes the provisions of that note for the affected instrumentation consistent with the revised APPLICABILITY of TS 3.7.2. These changes involve no physical changes to the facility and do not adversely affect the availability of the safety functions assumed for the MSIVs 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 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 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, as well as the proposed change to the APPLICABILITY for the MSIV TS (and the corresponding change to the footnote for the ESFAS Instrumentation in TS 3.3.2) 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.

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.

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

Date of amendment request: December 28, 2007.

Description of amendment request: The amendment would incorporate changes in the Technical Specifications (TS). Specifically, a footnote associated with Table 3.3.2–1 of Technical Specification 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' would be revised to make the exception allowed by the footnote consistent with the scope and Applicability of TS 3.7.3, ''Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and Main Feedwater Regulating Valve Bypass Valves (MFRVBVs)'' and a Note connected with each of two Surveillance Requirements (SRs), i.e., SR 3.7.2.1 and SR 3.7.2.2 under TS 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' would be deleted as it is no longer needed or appropriate for the affected SRs.

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 changes will not change accident initiators or precursors assumed or postulated in the final safety analysis report (FSAR)-described accident analyses, nor will they alter the design assumptions, conditions, and configuration 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 functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed changes do not physically alter safety-related systems, nor do they affect the way in which safety-related systems perform their 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. The applicable radiological dose acceptance criteria 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 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 (FQ), nuclear enthalpy rise hot channel factor (FAH), 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 surveillance 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, NW., Washington, DC 20037.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: November 29, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.4.10, ''Pressurizer Safety Valves,'' and TS 3.4.11, ''Pressurizer Power Operated Relief Valves (PORVs),'' to modify the completion times for default conditions in both TSs and to allow separate condition entry for PORV block valves in TS 3.4.11. The amendment request is adopting the following two Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) travelers to the standard TSs: TSTF–247–A and TSTF–352–A.

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.

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 the design and operating temperature and pressure limits placed on the reactor coolant system.

The proposed changes will not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration 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 functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their 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 plant]. The applicable radiological dose acceptance criteria 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. [Do] the proposed change[s] 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 SSC performs its 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. [Do] the proposed change[s] 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 (FQ), nuclear enthalpy rise hot channel factor (FΔH), 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 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.

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.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: February 13, 2008.

Brief description of amendment request: The amendments propose a one time steam generator (SG) tubing eddy current inspection interval revision to the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle 1 and 2) Technical Specifications (TSs) 5.5.9, ''Steam Generator (SG) Program,'' to incorporate an interim alternate repair criterion

(ARC) in the provisions for SG tube repair criteria during the Vogtle 1 inspection performed in refueling outage 14 and subsequent operating cycle, and during the Vogtle 2 inspection performed in refueling outage 13 and subsequent 18-month SG tubing eddy current inspection interval and subsequent 36-month SG tubing eddy current inspection interval. The amendments also revise TS 5.6.10, ''Steam Generator Tube Inspection Report,'' where three new reporting requirements are proposed to be added to the existing seven requirements.

*Date of publication of individual notice in **Federal Register:*** February 26, 2008 (73 FR 10305).

Expiration date of individual notice: April 28, 2008.

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.*

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: March 28, 2007, as supplemented by letter dated October 24, 2007.

Brief description of amendment: The amendment revised the required wattage specified in the River Bend Station, Unit 1, Technical Specification 5.5.7.e, Ventilation Filter Testing Program, for the Control Room Fresh Air System (CRFAS) heater for testing. The required wattage for testing the CRFAS heater was revised from 23 ± 2.3 kilowatt (kW) to ''$\geq$=15 kW.''

Date of issuance: February 28, 2008

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

Amendment No.: 159

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

*Date of initial notice in **Federal Register***: May 8, 2007 (72 FR 26175). The supplement dated October 24, 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** on May 8, 2007 (72 FR 26175).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2008.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 30, 2007, as supplemented by letter dated December 5, 2007.

Brief description of amendment: The amendment revised Technical Specification 3.1.3.4, ''Reactivity Control Systems CEA [Control Element Assembly] Drop Time,'' to change the individual rod drop time from the fully withdrawn position to 90 percent insertion from less than or equal to 3.5 seconds to less than or equal to 3.7 seconds.

Date of issuance: March 5, 2008.

Effective date: As of its date of issuance and shall be implemented prior to startup following the spring 2008 refueling outage.

Amendment No.: 275.

Renewed Facility Operating License No. NPF–6: The amendment revised the Technical Specifications and license.

*Date of initial notice in **Federal Register***: October 9, 2007 (72 FR 57354). The supplemental letter dated December 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 published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2008.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 15, 2007.

Brief description of amendment: The amendment changes Technical Specification (TS) Section 1.4 and Section 5. Changes to TS 1.4 incorporate Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Changes TSTF–284, ''Add 'Met vs. Perform' to Specification 1.4, Frequency,'' Revision 3, TSTF–485–A, ''Correction Example 1.4–1,'' Revision 0, and make administrative changes. Changes to TS Section 5 incorporate NRC-approved TSTF–258, ''Changes to Section 5.0, Administrative Controls,'' Revision 4, NRC-approved TSTF–273, ''[Safety Functions Determination Program] SFDP Clarifications,'' Revision 2, as amended by Westinghouse Owners Group (WOG) editorial change WOG–ED–23, and make administrative changes.

Date of issuance: March 5, 2008.

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

Amendment No.: 231

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications and Renewed License.

Date of initial notice in ***Federal Register:*** June 19, 2007 (72 FR 33782).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2008.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 18, 2007.

Brief description of amendments: The amendment revised Technical Specification 3.7.5, ''Control Room Area Ventilation Air Conditioning (AC) System,'' to add an Action Statement for two inoperable control room area ventilation AC subsystems. This operating license improvement was made available by the Nuclear Regulatory Commission on March 26, 2007 (72 FR 14143) as part of the consolidated line item improvement process.

Date of issuance: March 10, 2008

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

Amendment Nos.: 188/175

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications and License.

Date of initial notice in ***Federal Register:*** September 1, 2007 (72 FR 51860). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 10, 2008.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 17, 2006, as supplemented by letters dated September 21, 2007, December 21, 2007, February 1, 2008, and February 14, 2008.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement 3.3.1.1.8 to increase the frequency interval between Local Power Range Monitor (LPRM) calibrations from 1000 megawatt days per ton (MWD/T) average core exposure to 2000 MWD/T average core exposure. The LPRM system provides signals to associated nuclear instrumentation systems that serve to detect conditions in the core that have the potential to threaten the overall integrity of the fuel barrier. The LPRM system also incorporates features designed to diagnose and display various system trip and inoperative conditions.

Date of issuance: February 29, 2008.

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

Amendment Nos.: 266 and 270

Facility Operating License Nos. DPR–44 and DPR–56: Amendment revised the License and Technical Specifications.

Date of initial notice in ***Federal Register:*** August 28, 2007 (72 FR 49577). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2008.

No significant hazards consideration comments received: No.

FPL Energy, Point Beach, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: October 12, 2007, as supplemented by letters dated December 12, and December 21, 2007.

Brief description of amendments: The amendments revises Technical Specification 5.5.15 ''Containment Leakage Rate Testing Program,'' for Units 1 and 2. The proposed change allows a one-time interval extension of no more than 5 years for the Type A, Integrated Leakage Rate Test.

Date of issuance: February 26, 2008

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

Amendment Nos.: 232, 237

Renewed Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications/License.

Date of initial notice in ***Federal Register:*** December 4, 2007 (72 FR 68217). The supplements contained clarifying information and did not change the 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 February 26, 2008.

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

Brief description of amendments: The amendments revised Action Q of Technical Specifications Section 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' to reflect deletion of the power range neutron flux high negative rate trip function previously approved by Amendment Nos. 293 (for Unit 1) and 275 (for Unit 2).

Date of issuance: March 5, 2008

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

Amendment No.: 302 (for DCCNP–1) and 285 (for DCCNP–2)

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Renewed Operating Licenses and Technical Specifications.

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

The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 5, 2008.

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: May 22, 2007, as supplemented by letter dated December 5, 2007.

Brief description of amendments: The amendments revised the Technical Requirements Surveillance 13.3.33.2, Cycling Frequency for the Turbine Stop and Control Valves. The change will increase the valve cycle frequency interval from 12 to 26 weeks.

Date of issuance: February 29, 2008.

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

Amendment Nos.: Unit 1–143; Unit 2–143

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:*** August 14, 2007 (72 FR 45462). The supplement dated December 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 published in the **Federal Register** on August 14, 2007 (72 FR 45462).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 29, 2008.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 30, 2007, as supplemented by letters dated October 16, 2007, and November 2, 2007.

Brief description of amendment: The amendment changes the NMP2 Technical Specifications to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Analysis (ARTS/MELLLA). The Average Power Range Monitor (APRM) flow-biased simulated thermal power allowable value (AV) would be revised to permit operation in the MELLLA region. The current flow-biased Rod Block Monitor (RBM) would be replaced by a power dependent RBM, which also would require new AVs. The flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow dependent thermal limits administration. The Surveillance Requirement for the standby liquid control (SLC) system would be revised to require each SLC pump to deliver required flow at a discharge pressure ≥1325 psig in lieu of ≥1320 psig; the SLC relief valve setpoint would be increased from 1394 psig to 1400 psig. Finally, the proposed amendment employs a new model for performing the anticipated transients without scram analysis for ARTS/MELLLA conditions.

Date of issuance: February 27, 2008

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

Amendment No.: 123

Renewed Facility Operating License No. NPF–69: Amendment revises the License and Technical Specifications.

Date of initial notice in ***Federal Register:*** May 22, 2007 (72 FR 28721). The supplements dated October 16, 2007, and November 2, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2008.

No significant hazards consideration comments received: No

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: January 30, 2007, as supplemented by letter dated December 28, 2007.

Brief description of amendment: The amendment revised Technical Specifications (TSs) Surveillance Requirement (SR) 3.5.1.3.b to correctly state that the required pressure at which the Alternate Nitrogen System is determined to be operable should be greater than or equal to 410 psig, not the former stated pressure of greater than or equal to 220 psig. The safety-related Alternate Nitrogen System provides an alternate pressure source to equipment required during or following an accident. The licensee determined that the former acceptance value specified by SR 3.5.1.3.b (greater than or equal to 220 psig) was non-conservative and needed to be corrected to the higher value.

Date of issuance: February 21, 2008

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

Amendment No.: 155

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications and the Operating License.

Date of initial notice in ***Federal Register:*** March 27, 2007 (72 FR 14307). The supplemental letter 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 February 21, 2008.

No significant hazards consideration comments received: No

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California (TAC. No. J52690)

Date of application for amendment: May 17, 2006, supplemented January 25, 2008.

Brief description of amendment: The amendment approves a proposed change to the Physical Security Plan related to security post manning requirements.

Date of issuance: February 27, 2008

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

Amendment No.: 42

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

Date of initial notice in ***Federal Register:*** February 13, 2007 (72 FR 6788).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2008.

No significant hazards consideration comments received: No

PSEG Nuclear LLC, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: October 17, 2007, as supplemented on January 11, 2008.

Brief description of amendment: The amendment allows a one-time revision to the requirements for fuel decay time prior to commencing movement of irradiated fuel in the reactor. Specifically, the proposed amendment revises Technical Specification (TS) 3/4.9.3 to allow fuel movement to commence at 86 hours after the reactor is subcritical. The proposed change is only applicable to Salem Unit 2 refueling outage 2R16 which is scheduled to commence on March 11, 2008.

Date of issuance: March 5, 2008

Effective date: As of the date of issuance, to be implemented within 7 days.

Amendment No.: 271

Facility Operating License No. DPR–75: The amendment revises the TSs and the license.

Date of initial notice in ***Federal Register:*** December 4, 2007 (72 FR 68218). The letter dated January 11, 2008, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2008.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: August 28, 2007, as supplemented on October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008.

Brief description of amendments: The amendments revised the ''Maximum Power Level'' in paragraph 2.C(1) of the Vogtle Electric Generating Plant, Facility Operating Licenses NPF–68 and NPF–81 for Unit 1 and Unit 2, respectively. In addition, the amendments revised the definition of

''Rated Thermal Power (RTP)'' in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The proposed change increased the RTP from 3565 MWt to 3625.6 MWt, resulting in an increase of 1.7% from the current reactor output. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate.

Date of issuance: February 27, 2008

Effective date: As of the date of issuance and shall be implemented at the completion of spring 2008 refueling outage for Unit 1 and fall 2008 refueling outage for Unit 2.

Amendment Nos.: 149, 129

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register:*** November 20, 2007 (72 FR 65372). The supplements dated October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 2008.

No significant hazards consideration comments received: No

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: August 28, 2007, as supplemented on October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008.

Brief description of amendments: The amendments revised the ''Maximum Power Level'' in paragraph 2.C(1) of the Vogtle Electric Generating Plant, Facility Operating Licenses NPF–68 and NPF–81 for Unit 1 and Unit 2, respectively. In addition, the amendments revised the definition of ''Rated Thermal Power (RTP)'' in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The proposed change increased the RTP from 3565 MWt to 3625.6 MWt, resulting in an increase of 1.7% from the current reactor output. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate.

Date of issuance: February 27, 2008

Effective date: As of the date of issuance and shall be implemented at the completion of spring 2008 refueling outage for Unit 1 and fall 2008 refueling outage for Unit 2.

Amendment Nos.: 149, 129

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register:*** November 20, 2007 (72 FR 65372). The supplements dated October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 2008.

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: March 22, 2007, as supplemented by letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 2007.

Brief description of amendments: The amendments revised the licensing basis, pursuant to Title 10 of the *Code of Federal Regulations,* Section 50.67, ''Accident Source Term,'' and approved the methodology for evaluating radiological consequences of design-basis accidents as described in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents (DBAs) at Nuclear Power Reactors.'' The amendments revised the Technical Specifications in support of the revisions to the licensing basis.

Date of issuance: March 6, 2008

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

Amendment Nos.: Unit 1—182; Unit 2—169

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:*** July 31, 2007 (72 FR 41788). The supplemental letters dated April 10, July 18, October 11, November 13, December 13, and December 18, 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 March 6, 2008.

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 14, 2007, as supplemented by letter dated December 18, 2007.

Brief description of amendment: The amendment revised TS Table 3.3.2–1, ''Engineered Safety Features Actuation System Instrumentation,'' to separate the automatic actuation logic and actuation relays for steam line isolation (Function 4) and main feedwater isolation (Function 5) into the solid state protection system function and the main steam and feedwater isolation system. There are other proposed changes to the TSs and the plant in the application that are not being addressed in this amendment. The amendment to revise Surveillance Requirements 3.7.2.1 and 3.7.3.1 to replace the valve isolation times with the phrase ''within limits'' was issued August 28, 2007. The remaining TS and plant changes in the application will be addressed in future letters to the licensee.

Date of issuance: March 3, 2008

Effective date: As of its date of issuance and shall be implemented prior to the startup from Refueling Outage 16, scheduled for the spring of 2008.

Amendment No.: 175

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

Date of initial notice in ***Federal Register:*** The supplemental letter dated December 18, 2007, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the **Federal Register** on June 19, 2007 (72 FR 33785).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 3, 2008.

No significant hazards consideration comments received: No

Dated at Rockville, Maryland, this 17th day of March 2008.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E8–5734 Filed 3–24–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Office of New Reactors; Interim Staff Guidance on the Use of the GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents to Support Design; Certification and Combined License Applications; Solicitation of Public Comment

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Solicitation of public comment.

SUMMARY: The NRC is soliciting public comment on its Proposed Interim Staff Guidance COL/DC–ISG–005. This interim staff guidance supplements the guidance provided to the staff in Chapter 11, ''Radioactive Waste Management,'' of NUREG–0800, ''Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants,'' concerning the review of radioactive releases in gaseous and liquid effluents (GALE) to support design certification and combined license applications. This guidance provides a clarification on the use of a newer version of the boiling-water reactor and pressurized-water reactors GALE codes that is not referenced in the current NRC guidance. Upon receiving public comments, the NRC staff will evaluate and disposition the comments, as appropriate. Once the NRC staff completes the COL/DC–ISG–005, it will be issued for NRC and industry use. The NRC staff will also incorporate the approved COL/DC–ISG–005 into the next revision of the SRP and related guidance documents.

DATES: Comments must be filed no later than 30 days from the date of publication of this notice in the **Federal Register**. Comments received after this date will be considered, if it is practical to do so, but the Commission is able to ensure consideration only for comments received on or before this date.

ADDRESSES: Comments may be submitted to: Chief, Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC, 20555–0001.

Comments should be delivered to: 11545 Rockville Pike, Rockville, Maryland, Room T–6D59, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Persons may also provide comments via e-mail to Timothy Frye at *tjf@nrc.gov*. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *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 Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by e-mail at *pdr@nrc.gov*.

FOR FURTHER INFORMATION CONTACT: Mr. Timothy Frye, Chief, Health Physics Branch, Division of Construction, Inspection & Operational Programs, Office of the New Reactors, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone 301–415–3900 or e-mail at *tjf@nrc.gov*.

SUPPLEMENTARY INFORMATION: The agency posts its issued staff guidance in the agency external Web page (*http://www.nrc.gov/reading-rm/doc-collections/isg/*).

The NRC staff is issuing this notice to solicit public comments on the proposed COL/DC–ISG–005. After the NRC staff considers any public comments, it will make a determination regarding the proposed COL/DC–ISG–005.

Dated at Rockville, Maryland, this 19th day of March 2008.

For the Nuclear Regulatory Commission.

William D. Reckley,

Branch Chief, Rulemaking, Guidance and Advanced Reactors Branch, Division of New Reactor Licensing, Office of New Reactors.

[FR Doc. E8–5962 Filed 3–24–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Nuclear Waste and Materials; Meeting Notice

The Advisory Committee on Nuclear Waste and Materials (ACNW&M) will hold its 188th meeting on April 8–10, 2008, at 11545 Rockville Pike, Rockville, Maryland.

Tuesday, April 8, 2008, Room T–2B3

8 a.m.–4:10 p.m.: Working Group on the Effects of Low Radiation Doses Science And Policy (Open)—Purpose: The objectives of this Working Group Meeting are: (1) To discuss the Linear Non-Threshold (LNT) theory in light of current health physics, medical theory and cohort databases; (2) to review uncertainties about the presence or absence of health effects at low doses; (3) to examine the balance of science and policy in regulatory practice; (4) to discuss possible alternative approaches to the LNT theory in regulatory practice; and (5) to develop the information necessary to provide a letter report to the Commission.

8–8:05 a.m.: Greetings and Introductions (Open)—Dr. Michael Ryan, the cognizant ACNW&M Member for this meeting topic, will provide an overview of the expected goals for the Working Group Meeting, the planned technical sessions, and introduce the invited speakers.

8:05–8:25 a.m.: Opening Remarks by NRC Commissioner Peter B. Lyons (Open)

8:25 a.m.–4:10 p.m.: Session I: The State of the Science (Open)—This session will include six presentations. There will be a lunch break from 11:45 a.m.–1 p.m.

4:10–5 p.m.: Discussion of ACNW&M Letter Reports (Open)—The Committee will discuss potential ACNW&M letter reports on matters considered during previous meetings: (1) Managing Low-Activity Radioactive Waste; (2) Use of Burnup Credit for Licensing Spent Fuel Transportation Casks.

Wednesday, April 9, 2008, Room T–2B3

8:30 a.m.–4:10 p.m.: Working Group on the Effects of Low Radiation Doses Science and Policy—Continuation (Open)—Session II: Balancing Science and Policy in the Regulatory Area. There will be three presentations and a panel discussion. A lunch break will be held from 11:15 a.m.–1 p.m.

4:10–5 p.m.: Discussion of ACNW&M Letter Reports (Open)—Continued discussion of proposed and potential ACNW&M letter reports mentioned previously, as well as (3) Effects of Low Radiation Doses.

Thursday, April 10, 2008, Room T–2B1

8:30–8:35 a.m.: Opening Remarks by the ACNW&M Chairman (Open) The Chairman will make opening remarks regarding the conduct of today's sessions.

8:35 a.m.–12 p.m.: Discussion of ACNW&M Letter Reports (Open) (All) Continued discussion of proposed and potential ACNW&M letter reports previously listed.

4:10–5 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of ACNW&M activities and specific issues that were not completed during previous meetings. Discussions may include