7590-01-P

NUCLEAR REGULATORY COMMISSION

NRC-2009-0456

BIWEEKLY NOTICE

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 24, 2009, to October 7, 2009. The last biweekly notice was published on October 6, 2009 (74 FR 51327).

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

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

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

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

Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking and Directives Branch (RDB), TWB-05-B01M, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this *Federal Register* notice. Written comments may also be faxed to the RDB at 301-492-3446. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/doc-collections/cfr/. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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

To comply with the procedural requirements of E-Filing, at least ten (10) days prior to the filing deadline, the petitioner/requestor should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by calling (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the

Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate is available on NRC's public 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 using the agency's adjudicatory e-filing system may seek assistance through the "Contact Us" link located on the NRC website at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC Meta-System Help Desk, which is available between 8:00 a.m. and 8:00 p.m., Eastern Time, Monday through Friday, excluding government holidays. The Meta-System Help Desk can be contacted by telephone at 1-866-672-7640 or by e-mail at MSHD.Resource@nrc.gov.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

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

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

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

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

Date of amendment request: August 10, 2009.

Description of amendment request: The proposed amendment would revise the RBS Technical Specifications (TSs) to support operation with 24-month fuel cycles. Specifically, the change addresses certain TS Surveillance Requirement (SR) frequencies that are specified as "18 months" by revising them to "24 months" in accordance with the guidance of U.S. Nuclear Regulatory Commission (NRC) Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

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 TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle

length. The proposed TS changes do not physically impact the plant. The proposed TS changes do not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. The proposed TS changes do not impact the usefulness of the SRs in evaluating the operability of required systems and components, or the way in which the surveillances are performed. In addition, the frequency of surveillance testing is not considered an initiator of any analyzed accident, nor does a revision to the frequency introduce any accident initiators. The specific value of the allowable value is not considered an initiator of any analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously evaluated accident are not significantly increased. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Evaluation of the proposed TS changes demonstrated that the availability of credited equipment is not significantly affected because of other more frequent testing that is performed, the availability of redundant systems and equipment, and the high reliability of the equipment. Historical review of surveillance test results and associated maintenance records did not find evidence of failures that would invalidate the above conclusions.

The allowable values have been developed in accordance with [NRC Regulatory Guide] 1.105, “Instrument Setpoints,” to ensure that the design and safety analysis limits are satisfied. The methodology used for the development of the allowable values ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses and that the results and radiological consequences described in the safety analyses remain bounding. Therefore, the proposed change does not alter the ability to detect and mitigate events and, as such, does not involve a significant increase in the consequences of an accident previously evaluated.

<u>Standby Liquid Control System</u>

The proposed change in required weight of Boron-10 in [standby liquid control (SLC)] does not physically impact the plant, nor does it degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. The consequences of a previously evaluated accident are not increased. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Evaluation of the proposed TS changes demonstrated that the availability of credited equipment is not affected. Therefore, the proposed change does not alter the ability to detect and mitigate events and, as such, does not involve a significant increase in the consequences of an accident previously evaluated.

Loss of Power Instrumentation

A change to the Allowable Values (AVs) is proposed for Table 3.3.8.1-1, Item 1.c and Item 2.c. The proposed change is the result of application of the RBS Instrument Setpoint Methodology using plant-specific drift values and incorporating margins available based on a revised off-site reliability study. Application of this methodology results in AVs that more accurately reflect total device accuracy, as well as that of test equipment and calculated drift between surveillances. The proposed change will not result in any hardware changes. The instrumentation is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to the proposed changes. The role of the instrumentation is in mitigating and thereby, limiting the consequences of accidents.

The AVs were developed to ensure the design and safety analysis limits are satisfied. The methodology used for the development of the AVs ensures that: (1) the affected instrumentation remains capable of mitigating design basis events as described in the safety analysis, and, (2) the results and radiological consequences described in the safety analysis remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change in the degraded voltage protection voltage AVs allows the protection scheme to function as originally designed. The proposed allowable values ensure that the Class 1E distribution system remains connected to the offsite power system when adequate offsite voltage is available and motor starting transients are considered.

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

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

Response: No.

The proposed TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle length. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure modes are being introduced. The way surveillance tests are performed remains unchanged. A historical review of surveillance test results and associated

maintenance records indicated there was no evidence of any failures that would invalidate the above conclusions.

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

Standby Liquid Control System

The proposed change to the required weight of Boron-10 in SLC does not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure modes are being introduced. The way surveillance tests are performed remains unchanged. A historical review of surveillance test results and associated maintenance records indicated there was no evidence of any failures that would invalidate the above conclusions.

Loss of Power Instrumentation

The proposed change in AVs is the result of application of the Instrument Setpoint Methodology using plant-specific drift values and does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based upon the fact that the method and manner of plant operation are unchanged.

The use of the proposed AVs does not impact safe operation of the plant in that the safety analysis limits are maintained. The proposed change in AVs involves no system additions. The AVs are revised to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoints may be changed. No additional failure mechanisms are introduced as a result of the changes to the allowable values. Since operational methods remain unchanged and the operating parameters were evaluated to maintain the plant within existing design basis criteria, no different type of failure or accident is created.

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

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

Response: No.

The proposed TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle length. The impact of these changes on system availability is not

significant, based on other more frequent testing that is performed, the existence of redundant systems and equipment, and overall system reliability. Evaluations have shown there is no evidence of time dependent failures that would impact the availability of the systems. The proposed changes do not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed changes in TS instrumentation allowable values are the result of application of the RBS setpoint methodology using plant specific drift values. The revised allowable values more accurately reflect total instrumentation loop accuracy including drift while continuing to protect any assumed analytical limit. The proposed changes do not result in any hardware changes or in any changes to the analytical limits assumed in accident analyses. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to these changes. The proposed changes do not significantly impact any safety analysis assumptions or results.

Standby Liquid Control System

The proposed change in required weight of Boron-10 in SLC is to facilitate a change in the operating cycle length. The proposed change does not result in any hardware changes or in any changes to the analytical limits assumed in accident analyses. Existing operating margin between plant conditions and actual plant setpoints is not reduced due to this change. The proposed change does not impact any safety analysis assumptions or results. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Loss of Power Instrumentation

The proposed protection voltage AVs are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient voltage is available to the required equipment. The proposed change does not involve a reduction in a margin of safety. The proposed change was developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: August 25, 2009.

Description of amendment request: The proposed amendment would allow for a one-time extension to the ten-year frequency for the next Palisades Nuclear Plant (PNP) containment Type A integrated leak rate test (ILRT) that is required by Technical Specification (TS) 5.5.14. The proposed change would permit the existing ILRT frequency to be extended from ten years to approximately 11.25 years.

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

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

Response: No.

The proposed exemption involves a one-time extension to the current interval for Type A containment testing. The current test interval of 120 months (10 years) would be extended on a one-time basis to no longer than approximately 135 months from the last Type A test. The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

This proposed extension is for the Type A containment leak rate tests only. The Type B and C containment leak rate tests would continue to be performed at the frequency currently required by the PNP TS. As documented in NUREG 1493, Type B and C tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. The PNP Type A test history supports this conclusion.

The integrity of the containment is subject to two types of failure mechanisms that can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extension does not involve a significant increase in the 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 revision to the TS involves a one-time extension to the current interval for Type A containment testing. The containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed TS change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to the TS involves a one-time extension to the current interval for Type A containment testing. The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained. The proposed change involves only the extension of the interval between Type A containment leak rate tests. The proposed surveillance interval extension is bounded by the 15-month extension currently authorized within NEI 94-01, Revision 0. Type B and C containment leak rate tests would continue to be performed at the frequency currently required by TS.

> Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI and the Maintenance Rule serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A test interval. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

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

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

NRC Acting Branch Chief: Peter Tam

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc.,

Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: August 26, 2009.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Section 6.5 that governs administrative controls of High Radiation Areas (HRA) to incorporate the HRA administrative controls contained within the Standard Technical Specifications, NUREG-1433, Revision 3.

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

1. The operation of Vermont Yankee Nuclear Power Station (VY) in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed amendment does not impact the operability of any structure, system or component that affects the probability of an accident or that supports mitigation of an accident previously evaluated. The proposed amendment does not affect reactor operations or accident analysis and has no radiological consequences. The operability requirements for accident mitigation systems remain consistent with the licensing and design basis. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of VY in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed amendment does not change the design or function of any component or system. No new modes of failure or initiating events are being introduced. Therefore, operation of VY in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of VY in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The proposed amendment does not change the design or function of any component or system. The proposed amendment does not involve any safety limits, safety settings or safety margins. The TS administrative access controls for high radiation areas are being replaced with those contained in section 5.7 of NUREG-1433 to provide additional requirements and options for the control of these areas.

   Therefore, operation of VY in accordance with the proposed amendment will not involve a significant reduction in the margin to 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 C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 400 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Nancy Salgado.

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50-410, Nine Mile Point Nuclear Station Unit No. 2 (NMP 2), Oswego County, New York

Date of amendment request: May 27, 2009, as supplemented on August 28, 2009.

Description of amendment request: The proposed amendment requests an increase in the maximum steady-state power level at NMP2 from 3467 megawatts thermal (MWt) to 3988 MWt. This represents a 15-percent increase over the current licensed thermal power.

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

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

   No, the increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

   The proposed change will increase NMP2's authorized maximum power level from the current licensed thermal power (CLTP) level of 3467 megawatts thermal (MWt) to 3988 MWt. In support of this Constant Pressure Extended Power Uprate (CPPU), a comprehensive evaluation was performed for nuclear steam supply system (NSSS) and balance of plant (BOP) systems, structures, components, and analyses that could be affected by this change. The effect of increasing the maximum power level from the CLTP of 3467 MWt to 3988 MWt on the NMP2 licensing and design bases was evaluated. The result of this evaluation is that all plant components, as modified, will continue to be capable of performing their design function at an uprated core power of 3988 MWt. In addition, an evaluation of the accident analyses concludes that applicable analysis acceptance criteria continue to be met. Power level is an input assumption to the equipment design and accident analyses, but it is not an initiator for any transient or accident. Therefore, no accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change.

   Therefore, operation of the facility in accordance with the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   This change does not affect the release paths, the frequency of release, or the source term for release for any accidents previously evaluated in the Updated

Safety Analysis Report (USAR). Structures, systems, and components (SSC) required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition. The results of EPU [extended power uprate] accident evaluations do not exceed the U. S. Nuclear Regulatory Commission (NRC) approved acceptance limits.

The spectrum of postulated accidents and transients has been investigated and are shown to meet the regulatory criteria to which NMP2 is currently licensed. In the area of fuel and core design, the Safety Limit Minimum Critical Power ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDLS) are still met. Continued compliance with the SLMCPR and other SAFDLs is confirmed on a cycle specific basis consistent with criteria accepted by the NRC.

Challenges to the reactor coolant pressure boundary were evaluated at EPU conditions (pressure, temperature, flow, and radiation) and found to meet the acceptance criteria for allowable stresses. Adequate overpressure margin is maintained.

Challenges to the containment have been evaluated and the containment and its associated cooling system continue to meet applicable regulatory requirements. The increase in the calculated post Loss of Coolant Accident (LOCA) suppression pool temperature above the current peak temperature was evaluated and determined to be acceptable.

Radiological release events (accidents) have been evaluated and shown to meet the requirements of 10 CFR 50.67.

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

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

No, the increase in power level discussed herein will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will increase NMP2's authorized maximum power level from the CLTP level of 3467 MWt to 3988 MWt. Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. This Constant Pressure Extended Power Uprate utilizes a standard evaluation methodology applied to known technology employed within the range of current or modified plant capabilities. As such, the plant safety-related equipment continues to operate in accordance with regulatory criteria. Evaluations were performed using NRC approved codes,

standards and methods. No new accidents or event precursors have been identified.

All structures, systems and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes do not adversely affect safety-related systems or components and do not challenge the performance or integrity of any safety-related system. This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than was previously evaluated. Operating at a core power level of 3988 MWt does not create any new accident initiators or precursors.

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?

No, the increase in power level discussed herein will not involve a significant reduction in a margin of safety.

Comprehensive analyses of the proposed changes have concluded that relevant design and safety acceptance criteria will be met without a significant reduction in margins of safety. The analyses supporting EPU have demonstrated that the NMP2 SSCs are capable of safely performing at EPU conditions. The analyses identified and defined the major input parameters to the NSSS, analyzed NSSS design transients, and evaluated the capabilities of the NSSS fluid systems, NSSS/BOP interfaces, NSSS control systems, and NSSS and BOP components, as appropriate. Radiological consequences of design basis events remain within regulatory limits and are not increased significantly. The analyses confirmed that NSSS and BOP SSCs are capable, some with modifications, of achieving EPU conditions without significant reduction in margins of safety.
Analyses have shown that the integrity of primary fission product barriers will not be significantly affected as a result of the power increase. Calculated loads on SSCs important to safety have been shown to remain within design allowables under EPU conditions for all design basis event categories. Plant response to transients and accidents do not result in exceeding acceptance criteria. As appropriate, the evaluations that demonstrate acceptability of EPU have been performed using methods that have either been reviewed and approved by the NRC staff, or that are in compliance with regulatory review guidance and standards established for maintaining adequate margins of safety. These evaluations demonstrate that there are no significant reductions in the margins of safety.

Maximum power level is one of the inherent inputs that determine the safe operating range defined by the accident analyses. The Technical Specifications ensure that NMP2 is operated within the bounds of the inputs and assumptions used in the accident analyses. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the

> Technical Specifications. The engineering reviews performed for the constant pressure extended power uprate confirm that the accident analyses criteria are met at the revised maximum allowable thermal power level of 3988 MWt, as well as at the rated thermal power (RTP) levels specified in the Facility Operating License and Technical Specifications. Therefore, the adequacy of the revised Facility Operating Licenses and Technical Specifications to maintain the plant in a safe operating range is also confirmed, and the increase in maximum allowable power level does not involve a significant decrease in a margin of safety.
>
> 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Nancy L. Salgado.

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50-410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: June 29, 2009, as supplemented on August 13, 2009.

Description of amendment request: The proposed amendment would revise the NMP2 Technical Specification (TS) 5.5.12 by replacing the reference to Regulatory Guide (RG) 1.163 with a reference to Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 2-A, as the implementation document used by NMPNS to develop the NMP2 performance-based leakage testing program in accordance with Option B of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. The proposed amendment would allow the next primary containment integrated leak rate test (ILRT) to be performed within 15 years from the last ILRT as opposed to the current 10-year interval, and would allow successive ILRTs to be performed at 15-year intervals.

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 amendment involves changes to the NMP2 10 CFR 50 Appendix J Testing Program Plan. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2, for development of the NMP2 performance-based leakage testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less [then] the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval from 10 years to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small, and the increase in the large early release frequency resulting from the proposed change was determined to be within the guidelines published in NRC RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMPNS has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

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

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI-94-01, Revision 2, for development of the NMP2 performance-based leakage testing

program, and establishes a 15 year interval for the performance of the primary containment ILRT. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors and initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

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 amendment adopts the NRC-accepted guidelines of NEI-94-01, Revision 2, for development of the NMP2 performance-based leakage testing program, and establishes a 15 year interval for the performance of the primary containment ILRT. The amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the 10 CFR 50 Appendix J Testing Program Plan, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant safety analyses is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, B, and C containment leakage tests will continue to be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 2.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by an ILRT. In addition, the on-line containment monitoring capability that is inherent to inerted boiling water reactor containments allows for the detection of gross containment leakage that may develop during power operation. This combination of factors ensures that evidence of containment structural degradation is identified in a timely manner. Furthermore, a risk assessment using the current NMP2 Probabilistic Risk Assessment model concluded that extending the ILRT test interval from 10 years to 15 years results in a very small change to the NMP2 risk profile.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Nancy L. Salgado.

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

Date of amendment request: September 15, 2009.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.3.2, in Appendix A to Facility Operating License Nos. NPF-2 and NPF-8 for the Joseph M. Farley Nuclear Plant Units 1 and 2, respectively. P-11 is an engineered safety feature actuation system (ESFAS) permissive/interlock which permits normal unit cooldown and depressurization without actuation of safety injection (SI) from low pressurizer pressure. P-12 is an ESFAS permissive/interlock which permits normal unit cooldown and depressurization without actuation of SI and main steam line isolation on the condition of low steam line pressure. Both P-11 and P-12 circuits use input from three protection channels. The current wording of Condition K in TS 3.3.2 states, “Two channels inoperable.” As a result, Condition K does not explicitly address the possible conditions of one channel or three channels inoperable, possibly creating a literal compliance issue. The proposed Condition K change from “Two channels inoperable” to “One or more channels inoperable” will resolve the current literal compliance issue. The change does not alter the current Condition K required action, it simply clarifies that the required action must be performed for one, two, or three P-11 or P-12 channels inoperable. In addition, an editorial change is proposed for TS 5.6.8 to correct the citation of a condition requiring a report for the post-accident monitoring instrumentation. The current TS 5.6.8 text states, “When a report is required by Condition B or G of LCO [limiting conditions for operation]

3.3.3...." The citation of Condition B is correct while Condition G does not currently exist for LCO 3.3.3; instead TS 5.6.8 should cite Condition F.

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

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

Response: No.

The proposed change to TS 3.3.2 does not significantly increase the probability or consequences of an accident previously evaluated in the FSAR. These interlocks do not directly initiate an accident. The consequences of accidents previously evaluated in the FSAR are not adversely affected by these changes because the changes are made to reflect the Improved Standard Technical Specifications and the interlocks are verified to be in the required state for the unit condition.

The proposed change to TS 5.6.8 corrects an editorial error and therefore does not significantly increase the probability or consequences of a previously evaluated accident.

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

Response: No.

The proposed change to TS 3.3.2 does not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenario, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed TS 3.3.2 change does not challenge the performance or integrity of any safety-related systems. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed change to TS 5.6.8 corrects an editorial error and therefore does not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Response: No.

The proposed change to TS 3.3.2 does not involve a significant reduction in a margin of safety. The proposed change is made to accurately reflect the format of the Improved Standard Technical Specifications. The actuation setpoints specified by the Technical Specifications and safety analysis limits assumed in the accident analysis are unchanged. The margin of safety associated with these trip setpoints and the safety analysis acceptance criteria is unchanged. Therefore, the proposed change to TS 3.3.2 will not significantly reduce the margin of safety as defined in the Technical Specifications.

The proposed change to TS 5.6.8 corrects an editorial error and therefore involves no significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Branch Chief: Jon H. Thompson, Acting.

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

Date of amendment request: July 27, 2009 (TS-465).

Description of amendment request: The proposed change is to eliminate Technical Specification (TS) surveillance requirement (SR) 3.6.1.3.11 and the requirement to perform water leak rate testing on the listed containment isolation valves. More specifically, the proposed change eliminates water local leak rate testing of valves in the Containment Leak Rate Program that are being tested to verify the combined leakage rate is within the limit that

ensures the suppression pool level is sufficient to keep lines that terminate below the water level for at least 30 days without additional make-up.

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.

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

   This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to the scope of water leak rate testing for the subject valves does not affect the probability of the design basis accidents. The valves will continue to be maintained in an operable state, and in their current design configuration. There is no correlation between the scope of the water leak rate testing and accident probability.

   TVA reviewed the postulated consequences of design basis events on primary containment isolation under the proposed change. The primary containment structure, including access openings, penetrations and the containment heat removal system, is designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate (2.0% per day), the peak accident pressure and temperature that could occur during any postulated LOCA [loss-of-coolant accident].

   For the purposes of considering the consequences of LOCAs under the proposed change, a single active failure of a CIV [containment isolation valve] or a passive failure of the closed system were reviewed, within the limits of the existing licensing basis. Under the existing licensing basis, a pipe rupture of seismically qualified ECCS [emergency core cooling system] piping does not have to be assumed concurrent with the LOCA, except if it is a consequence of the LOCA. Consequential failures can be eliminated, since a LOCA inside containment is separated from the ECCS piping by the containment structure. Consequential failures of the ECCS piping from LOCA's outside containment are outside the Appendix J design considerations, although they are adequately addressed through the redundancy and separation of the ECCS design. A single active failure of the CIV, under the LOCA condition, can be accommodated since the closed and filled system piping and the suppression pool water inventory remain as the leakage barriers. The ECCS passive failure criterion does require consideration of system leaks, but not pipe breaks, beyond the initiating LOCA. Pipe leakage, equivalent to the leakage from a valve or pump seal failure, should be considered at 24 hours or greater post-LOCA. The capability to make-up inventory to the suppression pool is adequate to ensure that postulated seat leakage and pipe leakage does not result in a condition that jeopardizes pool

level. Make-up capability exists to the suppression pool. Actions to make-up to the suppression pool are delineated in Emergency Operating Instructions.

Therefore, the proposal to eliminate the subject water leak rate tests does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The acceptability of the proposed change to the scope of water leak rate testing for the subject valves is based on maintaining the existing barriers to primary containment leakage, and ensuring that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. By meeting these dual objectives, the plant response to the design basis events will be unchanged, and no new accident scenarios will be encountered. These two objectives are related, in that, the suppression pool inventory creates a passive barrier to primary containment atmospheric leakage for valves associated with penetrations which are located below the minimum water level of the pool.

The proposed Technical Specification change does not alter the configuration of the subject containment isolation valves or their associated systems. The valves will continue to be tested and maintained to ensure their operability. The subject valves are all isolation valves associated with lines that penetrate the primary containment. For closed system valves, the redundant isolation boundary for each of the affected valves is the closed system associated with the valve. The closed system piping is verified via a 10 CFR 50 Appendix J Type A test. The integrity of the closed systems is also monitored and controlled via Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment."

The subject valves may be open, or change state, post-accident to support the design function of their associated ECCS systems (HPCI [high-pressure coolant injection], Core Spray, RHR [residual heat removal]), RCIC [reactor core isolation coolant] or RHR Sampling using the Post Accident Sampling System. The subject valves function as system valves during the periods when they are open or in an intermediate state, not as containment isolation valves. Reliance is placed on the suppression pool seal and the closed system piping to maintain the barrier between primary and secondary containment atmospheres.

Therefore, with the valve configuration and closed systems configuration unaffected by the proposed change, the existing barriers to primary containment atmospheric leakage are maintained, so long as the suppression pool level is ensured.

The suppression pool is designed and operated so that it is filled with water in accordance with Technical Specifications 3.6.2.2, "Suppression Pool Water

Level," and the associated Bases. As such, the supply of water in the suppression pool is assured for 30 days during all design basis, post-accident modes of operation. Water leak rate testing has historically been performed on valves associated with lines that connect to the suppression pool. The acceptance criteria for combined leakage from these penetrations is 72.79 cfh [cubic feet per hour]. This leakage rate is at a level which ensures the 30 day post-accident suppression pool level.

As mentioned above, the integrity of the closed system piping is verified via a 10 CFR 50 Appendix J Type A test and is monitored and controlled via Technical Specification 5.5.2. TS 5.5.2 establishes a program to monitor and control leakage from systems located outside containment that could contain highly radioactive fluids during a serious transient or accident. This program applies to the ECCS and RCIC systems affected by the proposed change and ensures that leakage into secondary containment via packing, flanges, seals, etc., is controlled. Leakage from these systems has been found to be very low, and well below the 20 gpm [gallons per minute] limit established for these systems. The proposed change is not expected to contribute to higher levels of system leakage. Normal operational monitoring of suppression pool level, operator rounds, housekeeping inspections, and system pressure testing further ensure external leakage is identified and minimized while suppression pool level is being maintained.

A review of water leak rate test data for the subject CIVs showed that the valves have had leakage rates within the acceptance criteria. Testing of the valves in accordance with ASME [American Society of Mechanical Engineers] Code requirements ensure valve operability.

Therefore, leakage past the CIVs is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist to make-up water to the suppression pool if necessary. Existing Emergency Operating Instructions require actions if suppression pool level is less than the required level. Thus, the level of the suppression pool is ensured, independent of the current CIV water leak rate testing requirement.

The proposed change to the scope of water leak rate testing for the subject valves maintains the existing barriers to primary containment leakage, and ensures that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. Therefore, the plant response to the design basis events is unchanged, and the proposal 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?

This change does not involve a significant reduction in a margin of safety.

As discussed in the responses to questions 1 and 2, the proposed change does not alter the plant response to existing accident scenarios, and does not introduce

new or different scenarios. So the margin of safety from a design basis accident standpoint is maintained.

Historically, the leakage rate through the subject valves has been determined in accordance with TS SR 3.6.1.3.11. This leakage rate has always been within the acceptance criteria. Quantifying leakage past the CIVs has been used to ensure that the suppression pool level is assured for 30 days post-accident. Under the proposed change, this leakage rate will not be quantified. In addition, closed system leakage is monitored and controlled by an existing Technical Specification program. Closed system leakage has been found to be very low on each of the units, and is currently well below the 20 gpm allowable. Therefore, leakage past the CIVs is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist, and is proceduralized, to make-up water to the suppression pool if necessary. Thus the current capability to maintain adequate suppression pool level for 30 days post-accident is assured under the proposed change.

Therefore the proposed change to the scope of water leak rate testing for the subject valves 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

PREVIOUSLY PUBLISHED NOTICES OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

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

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

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

Date of application for amendment: September 1, 2009.

Description of amendment request: Delay the date specified in License Amendments 234 and 229 for the implementation of the Boraflex Remedy in the spent fuel pools.

Date of publication of individual notice in the *Federal Register*: September 15, 2009 (74 FR 47278).

Expiration date of individual notice: October 15, 2009 (Public comments) and November 16, 2009 (Hearing requests).

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

Date of application for amendments: July 23, 2009.

Description of amendments request: Revise the scope of the inservice inspections required in the tubesheet regions of the steam generators.

Date of publication of individual notice in the *Federal Register*: August 28, 2009 (74 FR 44405).

Expiration date of individual notice: September 28, 2009 (Public comments) and October 27, 2009 (Hearing requests).

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at

the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

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

Date of application for amendment: May 28, 2009, as supplemented by letter dated August 3, 2009.

Brief description of amendment: The amendments eliminated working hour restrictions from Technical Specification (TS) 5.2.2 for Palo Verde Nuclear Generating Station, Units 1, 2, and 3, to support compliance with the revisions to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 26, "Fitness for Duty Programs," that became effective on March 31, 2008. The changes are consistent with the NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification change traveler, TSTF-511, Revision 0, "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26."

Date of issuance: September 30, 2009.

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

Amendment No.: Unit 1 - 175; Unit 2 - 175; Unit 3 - 175.

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

Date of initial notice in *Federal Register*: July 28, 2009 (74 FR 37247). The supplemental letter dated August 3, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 13, 2009.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.5.8, "Inservice Testing Program," by incorporating TS Task Force Traveler (TSTF) 479, "Changes to Reflect Revision of 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.55a," and TSTF-497, "Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less." Specifically, the amendments (1) replace references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI with the ASME Code for Operation and Maintenance of Nuclear Power Plants for inservice testing activities, and (2) applies the extension allowance of SR 3.0.2 to other normal and accelerated inservice testing frequencies of 2 years or less that were not included in the frequencies of the table listed in TS 5.5.8.a.

Date of issuance: September 28, 2009.

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

Amendment Nos.: 294 and 270.

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

Date of initial notice in FEDERAL REGISTER: July 14, 2009 (74 FR 34046).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated September 28, 2009.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 20, 2008, as supplemented by letter dated August 12, 2009.

Brief description of amendment: The amendment revised Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," to add a reference to an analytical method that will be used to determine the core operating limits. The change is needed to support the use of GE14 fuel during refueling outage 15 scheduled for the fall of 2009.

Date of issuance: September 29, 2009.

Effective date: As of the date of issuance and shall be implemented prior to Cycle 16 operation.

Amendment No.: 166.

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

Date of initial notice in *Federal Register*: January 23, 2009 (74 FR 4249). The supplemental letter dated August 12, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: September 17, 2008, as supplemented by letters dated February 26, June 30, and September 24, 2009.

Brief description of amendment: The amendment revised the Waterford 3 Technical Specifications (TSs) to take credit for soluble boron in Region 1 (cask storage pit) and Region 2 (spent fuel pool and refueling canal) fuel storage racks for the storage of both Standard and Next Generation Fuel assemblies. Two new TSs were added which included a surveillance that ensures the required boron concentration is maintained in the spent fuel storage racks and to reflect the results of the new criticality analysis.

Date of issuance: September 30, 2009.

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

Amendment No.: 223.

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

Date of initial notice in Federal Register: April 14, 2009 (74 FR 17228). The application dated September 17, 2008, contained an evaluation of the TS change in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and the licensee determined that the change involved no significant hazards consideration (NSHC). However, based on the discussions between the staff and the licensee, the licensee provided a revised NSHC in its supplemental

letter dated February 26, 2009. Based on the February 26, 2009, revised NSHC, the staff's proposed NSHC determination was published in the *Federal Register* on April 14, 2009. The supplemental letters dated June 30 and September 24, 2009, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the staff's 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 September 30, 2009.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit No. 2, Beaver County, Pennsylvania

Date of application for amendment: October 10, 2008, as supplemented by letters dated. June 16 and July 14, 2009.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.5 to allow an additional method of repair for steam generator (SG) tubes by installation of leak limiting Alloy 800 sleeves developed by Westinghouse and clarifies an existing reporting requirement in TS 5.6.6.2.4 concerning SG tube inspections.

Date of issuance: September 30, 2009.

Effective date: As of the date of issuance and shall be implemented prior to achieving Mode 4 during startup from the fall 2009 refueling outage.

Amendment No: 170

Facility Operating License No. NPF-73. Amendment revised the License and TSs.

Date of initial notice in FEDERAL REGISTER: February 17, 2009 (74 FR 7482).

The June 16 and July 14, 2009, supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Northern States Power Company - Minnesota, LLC, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: May 29, 2009.

Brief description of amendment: The amendment changes the Technical Specifications, revising the applicability for isolation of the Reactor Water Cleanup System on a Standby Liquid Control system initiation to align with the modes stated in Specification 3.1.7.

Date of issuance: September 28, 2009.

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

Amendment No.: 164.

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

Date of initial notice in FEDERAL REGISTER: July 28, 2009 (74 FR 37248).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 2009.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: May 19, 2009, as supplemented August 28, 2009 (three submittals) and September 11, 2009

Brief description of amendments: The amendments revised TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic SG inspections (establish alternate repair criteria). The amendments also revised TS 5.6.10, "Steam Generator Tube Inspection Report," to remove reference to previous interim alternate repair criteria and provide specific reporting requirements for Unit 1 during refueling outage (RFO) 15 and the subsequent operating cycle, and for Unit 2 during RFO 14 and the subsequent operating cycle.

Date of issuance: September 24, 2009.

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

Amendment Nos.: 157 and 138.

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

Date of initial notice in *FEDERAL REGISTER*: June 18, 2009 (74 FR 28962).

The supplements dated August 28, 2009, and September 11, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 24, 2009.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 4, 2009.

Brief description of amendment: The amendment revised the Callaway Plant Technical Specification (TS) 5.2.2, ‘‘Unit Staff,’’ to eliminate working hour restrictions in paragraph d of TS 5.2.2 to support compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26. The change is consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Revision 0 to TS Task Force (TSTF) Improved Technical Specification change traveler, TSTF-511, ‘‘Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26.’’ The availability of this TS improvement was announced in the *Federal Register* on December 30, 2008 (73 FR 79923), as part of the consolidated line item improvement process.

Date of issuance: September 29, 2009.

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

Amendment No.: 193.

Facility Operating License No. NPF-30: The amendment revised the Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: July 28, 2009 (74 FR 37250).

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

Date of initial notice in *Federal Register*: July 28, 2009 (74 FR 37250).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of October 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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