A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html.* The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions about the information collection requirements may be directed to the NRC Clearance Officer, Margaret A. Janney (T–5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, by telephone at 301–415–7245, or by e-mail to *INFOCOLLECTS@NRC.GOV.*

Dated at Rockville, Maryland, this 22nd day of January 2008.

For the Nuclear Regulatory Commission.

Gregory Trussell,

Acting NRC Clearance Officer, Office of Information Services.

[FR Doc. E8–1507 Filed 1–28–08; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 3, to January 16, 2008. The last biweekly notice was published on January 15, 2008 (73 FR 2546).

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™ 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 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*.

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

Date of amendment request: November 14, 2007.

Description of amendment request: The proposed amendments would modify the Technical Specifications (TS) by adding Limiting Condition for Operation (LCO) 3.0.8 on the inoperability of snubbers using the Consolidated Line Item Improvement Process (CLIIP). The proposed amendments would also make conforming changes to TS LCO 3.0.1. This request is consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler No. 372, Revision 4, ''Addition of LCO 3.0.8, Inoperability of Snubbers.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible license amendments adopting TSTF–372 using the NRC's 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 May 4, 2005 (70 FR 23252), which included the resolution of public comments on the model SE. The May 4, 2005, notice of availability referenced the November 24, 2004, notice. 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[s] [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change[s] [allow] a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore, the consequences of an accident previously evaluated are not significantly affected by [these] change[s]. The addition of a requirement to assess and manage the risk introduced by [these] change[s] will further minimize possible concerns. Therefore, [these] change[s] [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The proposed change[s] [do] not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Criterion 3—The proposed change[s] [do] not involve a significant reduction in the margin of safety.

The proposed change[s] [allow] a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in [NRC] RG [Regulatory Guide] 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk[, which is required by the proposed TS 3.0.8]. The net change to the margin of safety is insignificant. Therefore, [these] change[s] [do] 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: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072–2034

NRC Branch Chief: Thomas G. Hiltz.

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina.

Date of amendments request: September 29, 2007, as supplemented on December 7, 2007.

Description of amendments request: The amendment would revise the Technical Specification (TS) Administrative Controls section pertaining to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) requirements for inservice testing of pumps and valves. The changes are based on Technical Specification Task Force (TSTF) Traveler TSTF–479, ''Changes to Reflect Revision of 10 CFR 50.55a,'' as modified by TSTF–497, ''Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less.''

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

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

Response: No.

The proposed change revises TS 5.5.6, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed change does not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change revises TS 5.5.6, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or involve a 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 offsite and there is no increase in individual or cumulative occupational exposure. Therefore, the proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No.

The proposed change revises TS 5.5.6, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3. 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 the methods governing normal plant operation. 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 function of the affected pumps and valves will be maintained. 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Thomas H. Boyce.

Duke Power Company LLC, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina.

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: November 12, 2007.

Description of amendment request: The amendments would approve proposed changes to the licensing bases and final updated safety analysis report for both the Catawba Nuclear Power Station, Units 1 and 2, and the McGuire Nuclear Power Station, Units 1 and 2, concerning Revision 1 to DPC–NE–1005–P, Nuclear Design Methodology Using CASMO–4/SlMULATE–3 MOX.

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 UFSAR change to allow the use of the CASMO–4/SIMULATE–3 MOX reload design software to analyze reactor cores with fuel containing gadolinia does not involve a significant increase in the probability or consequences of an accident previously evaluated. The CASMO–4 and SIMULATE–3 MOX codes are used to perform reactivity and power distribution calculations to develop power distribution limits and provide confirmation of reactivity and power distribution input assumptions used in the evaluation of UFSAR Chapter 15 accidents. The SIMULATE–3 MOX code is also used to confirm the acceptability of thermal limits at post accident conditions. Since the CASMO–4/SIMULATE–3 MOX software is not used in the operation of any plant equipment, the probability of an accident previously evaluated in the UFSAR is not increased.

The benchmark calculations performed in Revision 1 to DPC–NE–1005–P verified the acceptability of the CASMO–4/SIMULATE–3 MOX codes for performing reload design calculations for reactor cores containing gadolinia. These calculations confirmed the accuracy of the codes and developed a methodology for calculating power distribution uncertainties for use in reload design calculations. The use of power distribution uncertainties applicable to gadolinia core designs in conjunction with predicted peaking factors ensures that thermal accident acceptance criteria are satisfied.

Therefore, the proposed amendment 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 extension of the reload design software to perform reload design calculations for reactor cores containing gadolinia will not create the possibility of a new or different kind of accident from any accident previously evaluated. The CASMO–4/SIMULATE–3 MOX software is not installed in any plant equipment and therefore the software is incapable of initiating an equipment malfunction that would result in a new or different type of accident from any previously evaluated. The evaluation of UFSAR accidents and the associated acceptance criteria for these accidents remains unchanged.

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 extension of the CASMO–4/SIMULATE–3 MOX reload design software to perform reload design calculations for reactor cores containing gadolinia will not involve a significant reduction in a margin of safety.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design function during and following an accident. These barriers include the fuel cladding, the reactor coolant system and the containment system. The reload design process assures the acceptability of thermal limits under normal, transient, and accident conditions. The CASMO–4/SIMULATE–3 MOX reload design software was qualified for the analysis of reactor cores containing gadolinia in Revision 1 to DPC–NE–1005–P and a methodology for developing appropriate power distribution uncertainties for application in reload design analyses was developed. The use of these uncertainties for analysis of reload cores with gadolinia ensures that design and safety limits are satisfied such that the fission product barriers perform their design function.

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: 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: John Stang, Acting.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts.

Date of amendment request: November 29, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to control room envelope habitability in TS 3.7.B.2 ''Control Room High Efficiency Air Filtration System (CRHEAFS)'' and TS Section 5.5 ''Administrative Controls—Programs and Manuals'' consistent with Technical Specification Task Force (TSTF)-448, Revision 3.

The availability of TS improvement was announced in the **Federal Register** on January 17, 2007 (72 FR 2022), including a model safety evaluation and model no significant hazards consideration (NSHC) determination, as part of the consolidated line item improvement process.

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 a 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 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: Mark G. Kowal.

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

Date of amendment request: January 2, 2008.

Description of amendment request: The proposed amendment revises the action requirements for certain inoperable containment isolation valves in Technical Specification 3/4.6.3, ''Containment Isolation Valves,'' to increase the allowed outage time from 4 hours to 72 hours.

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 modifies existing action requirements for inoperable containment isolation valves. The condition evaluated, the Action requirements and the associated allowed outage times do not impact initiating conditions for any accident previously evaluated. Containment integrity will continue to be maintained by the closed system when the proposed actions are implemented. The new action requirement provides appropriate remedial actions to be taken in response to an inoperable containment isolation valve in a closed system while minimizing the risk associated with continued operation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve any changes to plant equipment or system design functions. The specification for containment isolation valves provides controls for maintaining the containment pressure boundary. The new action requirement and surveillance requirement are sufficient to ensure that the containment isolation function is maintained. No new accident initiators are introduced by this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The new action requirement does not involve a significant reduction in the margin of safety. The proposed action for an inoperable containment isolation valve in a closed system minimizes the risk of continued operation under the specified conditions, considering the reliability of the closed system (i.e., passive barrier), a reasonable time for repairs or replacement of the isolation feature, and that 72 hours is typically provided for losing one train of redundancy throughout the NUREGs, and the low probability of a design basis accident occurring during the allowed outage time period (reference TSTF [Technical Specifications Task Force]-30). Should the penetration required to be isolated, Technical Specification 3.6.1.1 provides the surveillance requirement to verify at least once every 31 days that the affected penetration flow path is isolated if the penetration is not capable of being closed by operable containment isolation valves. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas G. Hiltz.

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa.

Date of amendment request: November 14, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow use of the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI, Subsection IWE for visual examination of the steel containment. This license amendment request is consistent with NRC approved Technical Specification Task Force (TSTF) Traveler number TSTF–343, Revision 1, ''Containment Structural Integrity.''

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

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

Response: No.

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class MC. The proposed change affects the frequency of visual examinations that will be performed for the containment. The frequency of visual examinations of the containment has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Section Xl Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained. The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility.

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

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

Response: No.

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class MC. The change affects the frequency of visual examinations that will be performed for the containment. 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 safety function of the containment as a fission product barrier is maintained. 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, 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 a margin of safety?

Response: No.

The proposed change revises the Improved Standard Technical Specification Administrative Controls program requirements for consistency with the requirements of 10 CFR 50, paragraph 55a(g)(4) for components classified as Code Class MC. The change affects the frequency of visual examinations that will be performed for the containment. The safety function of the containment as a fission product barrier will be maintained.

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: Marjan Mashhadi, Florida Power & Light Company, 801 Pennsylvania Avenue, Suite 220, Washington, DC 20004.

NRC Acting Branch Chief: Cliff Munson.

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 amendment request: December 29, 2007.

Description of amendment request: The amendment would revise the Point Beach Nuclear Plant (PBNP) Units 1 and 2 Technical Specifications (TS) requirement for the completion time (CT) of TS 3.7.5.C. This revision would allow two separate one-time extensions of the CT for TS 3.7.5.C from seven days to 16 days; one extension for each of the train-specific motor-driven auxiliary feedwater (MDAFW) pumps.

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

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

Response: No.

The results of the Technical Evaluation (Section 3.0) [of the application] demonstrate that, with the requested change, the increase in the probability of an accident previously evaluated fall within the guidance in RG 1.177 [Regulatory Guide 1.177, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications]. Therefore, the risk impact of the proposed CT extensions is small.

The ability of the AFW [auxiliary feedwater] system to deliver the required flow to mitigate design basis accidents is maintained. The ability to isolate AFW flow to or steam supply from the affected steam generator during design basis accidents is unaffected by this requested change. The applicable radiological analyses remain bounding.

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

Response: No.

The requested change to extend the CT of TS 3.7.5.C from 7 days to 16 days to replace a MDAFW pump and motor will not create the possibility of a new or different kind of accident. Two unit-specific TDAFW pump systems and one MDAFW pump system will remain OPERABLE and capable of performing the AFW system function. Prior to taking the MDAFW pump out of service for pump and motor replacement, both unit-specific turbine-driven auxiliary feedwater (TDAFW) pump systems and the other MDAFW pump system will be demonstrated OPERABLE. To ensure that the redundant AFW pump systems remain OPERABLE, risk management actions will be taken that include protecting the redundant operable AFW pump systems.

To manage the fire risk due to a MDAFW pump being inoperable, compensatory measures will be initiated to monitor and ensure that combustible loading, work activities, and other activities that could increase the likelihood of a fire are minimized. An initial baseline and weekly thermography of potential fire initiators will be performed to detect degrading operating equipment. No new failure will be created.

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

Response: No.

The ability of the AFW system to deliver the required flow to mitigate design basis accidents will be maintained. The ability to isolate AFW flow to or steam supply from the affected steam generator during design basis accidents is unaffected by this requested change. The applicable radiological analyses remain bounding. No significant reduction in a margin of safety will occur.

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: Antonio Fernandez, Esquire, Senior Attorney, FPL Energy Point Beach, LLC, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Acting Branch Chief: Cliff Munson.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio.

Date of amendment request: September 5, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 3.6.1, 3.6.4, and 3.6.5 to relax the position verification requirements for primary containment isolation devices, secondary containment isolation devices, and drywell isolation devices that are locked, sealed, or otherwise secured. These changes are based on TS Task Force (TSTF) change traveler TSTF–45 (Revision 2) and TSTF–269 (Revision 2), which have been approved generically for the Boiling Water Reactor (BWR) Standard Technical Specifications, NUREG–1434 (BWR/6).

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

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

Response: No.

The proposed change will revise the position verification requirements for manual containment and drywell isolation devices that are locked, sealed, or otherwise secured in the closed position. Revising the verification requirements will not introduce any physical changes or result in the equipment being operated in a new or different manner. All systems, structures, and components previously required for mitigation of a transient remain capable of performing their designed functions. Furthermore, although the proposed change would revise the position verification requirements, no physical change is being made to the assumed position of the valves for accident analysis. Therefore, this change does not involve a significant increase to the probability or consequences of an accident previously evaluated.

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

Response: No.

No new accident scenarios or failure mechanisms are introduced as a result of this proposed change. The proposed amendment would revise the position verification requirements but not alter any valve positions. With no changes to the plant lineup, no new or different accidents are possible. 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 a margin of safety?

Response: No.

The proposed amendment revises the position verification requirements for manual containment and drywell isolation valves that are locked, sealed, or otherwise secured in the closed position. The revised position verification requirements have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. Additionally, position verification does not alter the actual valve positions, introduce any physical changes, or reduce the ability of the valve to control leakage rates during design basis radiological accidents. 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: David W. Jenkins, Attorney, FirstEnergy Corporation, Mail Stop A–GO–15, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Russell Gibbs.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio.

Date of amendment request: September 18, 2007.

Description of amendment request: The proposed license amendment would modify technical specification (TS) requirements related to control room envelope habitability in accordance with Technical Specification Task Force (TSTF) Change Traveler TSTF–448, Revision 3, per the consolidated line item improvement process (CLIIP).

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075–61084), on possible amendments concerning the CLIIP, including a model safety evaluation and a model no significant hazards consideration 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), as part of the CLIIP. In its application dated September 18, 2007, the licensee affirmed the applicability of the following determination.

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 an 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 the analysis adopted by the licensee 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 amendment request involves no significant hazards consideration.

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, Mail Stop A–GO–15, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Russell Gibbs.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit 3 Nuclear Generating Plant (CR–3), Citrus County, Florida.

Date of amendment request: July 31, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to impose more restrictive voltage and frequency limits during surveillance testing of the emergency diesel generators.

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

The LAR [license amendment request] proposes to provide more restrictive steady state voltage and frequency limits for the Emergency Diesel Generators (EDGs). The voltage band is going from a range of greater than or equal to 3933 VAC [volts, alternating current] but less than or equal to 4400 VAC to greater than or equal to 4077 VAC but less than or equal to 4243 VAC. The proposed limits are plus or minus 2% around the nominal safety-related bus voltage of 4160 VAC. The Frequency Limits are going from a 2% tolerance band to a 1% tolerance band around the nominal frequency of 60 Hz (59.4 to 60.6 Hz), for fast starts and emergency starts of the EDGs. These acceptance limits are specifically for steady state conditions following a fast start of the EDGs.

Slow starts will also have a more restrictive frequency band, but it will be slightly larger than for fast starts. The reason for this difference is based on the speed control circuitry for the EDG. The EDG has an electro-mechanical component in the slow start circuitry that is not present in the fast start circuitry. The proposed slow start limits are plus or minus 1.5% (59.1 Hz to 60.9 Hz). The voltage limits for a slow start will be the same as for a fast start.

The EDGs are a safety related system that functions to mitigate the impact of an accident with a concurrent loss of offsite power. A loss of offsite power is typically a significant contributor to postulated plant risk and, as such, onsite AC generators have to be maintained available and reliable in the event of a loss of offsite power event. The EDGs are not initiators for any analyzed accident, therefore; the probability for an accident that was previously evaluated is not increased by this change. The revised, voltage and frequency limits will ensure the EDGs will remain capable of performing their design function.

The consequences of an accident refer to the impact on both the plant personnel and the public from any radiological release associated with the accident. The EDG supports equipment that is supposed to preclude any radiological release. More restrictive voltage and frequency limits for the output of the EDG restores design margin, and provides assurance that the equipment supplied by the EDG will operate correctly and within the assumed timeframe to perform their mitigating functions.

Until the proposed CR–3 ITS [improved TS] EDG voltage and frequency limits are approved, administratively controlled limits have been established in accordance with Administrative Letter 98–10 to ensure all EDG mitigation functions will be performed in the event of a loss of offsite power. These administrative limits have been determined as acceptable and have been incorporated into the Surveillance test procedures under the provisions of 10 CFR 50.59. Periodic testing has been performed with acceptable results. Since EDGs are mitigating components and are not initiators for any analyzed accident, no increased probability of an accident can occur. Since administrative limits will ensure the EDGs will perform as designed, consequences will not be significantly affected.

(2) Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Administrative voltage limits were established using verified design calculations and the guidance of NRC Administrative Letter 98–10. These administrative limits will ensure the EDGs will perform as designed. No new configuration is established by this change. The administrative limits for the EDG frequency were determined to be sufficient to account for measurement and other uncertainties.

The proposed amendment will place the administrative limits into the CR–3 ITS. The more restrictive voltage and frequency limits will provide additional assurance that the EDG can provide the necessary power to supply the required safety-related loads during an analyzed accident.

The proposed voltage and frequency ITS limits restore the EDG capability to those analyzed. No new configuration is established. Therefore, no new or different kind of accident from any previously evaluated can be created.

(3) Does not involve a significant reduction in a margin of safety.

The LAR proposes to provide more restrictive steady state voltage and frequency limits for the EDGs. The change in the acceptance criteria for specific surveillance testing provides assurance that the EDGs will be capable of performing their design function. Previous test history has shown that the new limits are well within the capability of the EDGs and are repeatable. The frequency ''as left'' setting will be adjusted such that it remains within a tight band and this assures the ''as found'' setting will be in the acceptable band. The requirement to adjust the as left frequency setting as well as the limitations on the frequency as left tolerance have been proceduralized to assure the requirement is satisfied.

The proposed ITS limits on voltage and frequency will assure the EDG will be able to perform all design function assumed in the accident analyses. Administrative limits are in place to ensure these parameters remain within analyzed limits. As such, 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Thomas H. Boyce.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit 3 Nuclear Generating Plant, Citrus County, Florida.

Date of amendment request: October 25, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) by relocating references to specific American Society for Testing and Materials (ASTM) standards for fuel oil testing to licensee-controlled documents. The proposed change is based on TS Task Force (TSTF) Traveler TSTF–374, ''Revision to TS 5.5.13 and Associated Bases for Diesel Fuel Oil,'' and was submitted using the Consolidated Line Item Improvement Process (CLIIP). Some changes included in TSTF–374, such as the addition of alternate criteria to the ''clear and bright'' acceptance test for new fuel oil, were not included in the application because they are already part of the licensing basis for Crystal River Unit 3.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 22, 2006 (71 FR 9179), on possible amendments to revise plant-specific TSs in accordance with TSTF–374, including a model safety evaluation and model No Significant Hazards Consideration (NSHC) Determination, using the CLIIP. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 21, 2006 (71 FR 20735). The licensee affirmed the applicability of the model NSHC determination in its application dated October 25, 2007.

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

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

Response: No.

The proposed changes relocate the specific ASTM standard references from the Administrative Controls Section of TS to a licensee-controlled document. Requirements to perform testing in accordance with applicable ASTM standards are retained in the TS as are requirements to perform surveillances of both new and stored diesel fuel oil. Future changes to the licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, tests and experiments,'' to ensure that such changes do not result in more than a minimal increase in the probability or consequences of an accident previously evaluated. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to recognize more rigorous testing of water and sediment content. Relocating the specific ASTM standard references from the TS to a licensee-controlled document and allowing a water and sediment content test to be performed to establish the acceptability of new fuel oil will not affect nor degrade the ability of the emergency diesel generators (DGs) to perform their specified safety function. Fuel oil quality will continue to meet ASTM requirements.

The proposed changes do 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 do not adversely affect the ability of structures, systems, and components (SSCs) to perform their intended safety 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 any accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. Therefore, the changes do not involve a significant increase in the probability or consequences of any 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 changes relocate the specific ASTM standard references from the Administrative Controls Section of TS to a licensee-controlled document. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil. 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. The requirements retained in the TS continue to require testing of the diesel fuel oil to ensure the proper functioning of the DGs.

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

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

Response: No.

The proposed changes relocate the specific ASTM standard references from the Administrative Controls Section of TS to a licensee-controlled document. Instituting the proposed changes will continue to ensure the use of applicable ASTM standards to evaluate the quality of both new and stored fuel oil designated for use in the emergency DGs. Changes to the licensee-controlled document are performed in accordance with the provisions of 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that diesel fuel oil testing is conducted such that there is no significant reduction in a margin of safety.

The ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil. The margin of safety provided by the DGs is unaffected by the proposed changes since there continue to be TS requirements to ensure fuel oil is of the appropriate quality for emergency DG use. The proposed changes provide the flexibility needed to improve fuel oil sampling and analysis methodologies while maintaining sufficient controls to preserve the current margins of safety.

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

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

NRC Branch Chief: Thomas H. Boyce.

Indiana Michigan Power Company (I&M), Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1 (DCCNP–1), Berrien County, Michigan.

Date of amendment request: December 27, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) Section 3.4.1, ''RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,'' to increase the minimum reactor coolant system (RCS) flow rate from 341,100 to 354,000 gallons per minute. The new analysis is performed using the NRC-approved methodology set forth in Westinghouse Topical Report WCAP–16009–P–A, ''Realistic Large-Break LOCA [Loss-of-Coolant Accident] Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)''; the licensee proposed to endorse this methodology by a revision of Section 5.6.5, ''Core Operating Limits Report (COLR).''

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

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

No. The proposed amendment would revise the subject TS sections to endorse a change in licensing basis, which involves use of an NRC-approved large break LOCA analysis methodology as set forth in Topical Report WCAP–16009–P–A, and to increase the required RCS flow rate. This change in licensing basis does not result in modification of plant design or method of operation that could change initiators of previously analyzed accidents. Further, this change does not modify the design performance of structures, systems, and components, relied upon to mitigate previously analyzed accidents. Thus, DCCNP–1 will continue to operate as before, resulting in no significant increase of the probability of occurrence of any accident previously analyzed, and no significant increase in consequences should any of the previously analyzed accidents occur.

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

No. The proposed TS and licensing basis changes would support a modification permitting four-loop injection of the low-head safety injection system, an accident-mitigating system. Accident-mitigating systems are not identified as accident initiators in previously analyzed accidents. There is no modification of other structure, system, or component, and no change to reactor protection system or engineered safeguards feature actuating system setpoints. Accordingly, no new transient or accident event would result due to modification of the low-head safety injection system. In addition, employing the ASTRUM methodology in an analysis does not create any new failure modes that could lead to a different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the models and associated assumptions used to analysis the system's performance. The subject system will continue to perform the same accident-mitigating function to the same level of reliability as defined in the DCCNP–1 Updated Safety Analysis Report. The analysis model to be endorsed by the revised TS is an NRC-approved methodology which will continue to show that DCCNP–1 operates with the same margin of safety. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Kimberly A. Harshaw, Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Cliff Munson.

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

Date of amendment request: December 27, 2007.

Description of amendment request: The proposed amendment would modify Technical Specifications (TS) requirements related to control room envelope habitability in TS Section 3.7.10, ''Control Room Emergency Ventilation (CREV) System,'' and Section 5.5, ''Programs and Manuals.'' The proposed changes are consistent with Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) change TSTF–448, Revision 3.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC) by referencing the NRC staff's model NSHC analysis published on January 17, 2007 (72 FR 2022). The NRC staff's model NSHC analysis is reproduced 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 the licensee's referenced analysis, and has found that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the proposed amendment involves no significant hazards consideration.

Attorney for licensee: Kimberly A. Harshaw, Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Cliff Munson.

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

Date of amendment request: November 19, 2007.

Description of amendment request: The proposed changes to the license and Technical Specifications reflect an increase in the rated thermal power from 2381 to 2419 megawatts thermal

(1.62 percent increase) based upon increased feedwater flow measurement accuracy to be achieved by using high accuracy ultrasonic flow measurement instrumentation.

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

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

Response: No.

The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of components and systems that could be affected by this change. Evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable.

The primary loop components (reactor vessel, reactor intemals, control rod drive housings, piping and supports, recirculation pumps, etc.) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the Nuclear Steam Supply Systems (NSSS) will still perform their intended design functions during normal and accident conditions. The balance of plant (BOP) systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level.

Because the integrity of the plant will not be affected by operation at the uprated condition, NPPD [Nebraska Public Power District] has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the core thermal power uncertainty measurement allows a majority of the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 2419 MWt [mmegawatts thermal]. Other analyses performed at a nominal power level have either been evaluated or re-performed for the 1.62% increased power level. The results demonstrate that acceptance criteria of the applicable analyses continues to be met at the 1.62% uprate conditions. As such, all CNS [Cooper Nuclear Station] USAR [updated safety analysis report] Chapter 14 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.62% uprated condition.

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

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

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier, and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed or approved by the Nuclear Regulatory Commission, or that are in compliance with regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: Thomas G. Hiltz.

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket Nos. 50–220 and 50–410, Nine Mile Point Nuclear Station Unit Nos. 1 (NMP1) and 2 (NMP2), Oswego County, New York.

Date of amendment request: December 20, 2007.

Description of amendment request: The proposed amendment would revise NMPI Technical Specification (TS) 6.3, ''Unit Staff Qualifications,'' and NMP2 TS 5.3, ''Unit Staff Qualifications,'' to update requirements that have been superseded due to the accreditation of the NMPNS licensed operator training program and due to promulgation of the revised Title 10 of the Code of Federal Regulations (10 CFR), Part 55, ''Operators' Licenses,'' which became effective on May 26, 1987 (52 FR 9453). Additionally, the proposed amendment would revise NMP1 TS 6.3 by eliminating the qualification requirement exceptions listed for the position of Manager Operations, and previously approved by the Nuclear Regulatory Commission (NRC) staff. The position of Manager Operations will meet the minimum qualification requirements as required in American National Standard Institute (ANSI) Standard NI8.1–1971, ''American National Standard for Selection and Training of Nuclear Power Plant Personnel.''

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 Technical Specifications change to the licensed operator qualification requirements is an administrative change to revise the present operator qualification program to the more current National Academy for Nuclear Training (NANT) guidelines for initial training and qualification of licensed operators. The change conforms to the current requirements of 10 CFR [Part] 55, ''Operators' Licenses.''

Although the licensed operator qualification and training program may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR [Part] 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is accredited and is based on a systems approach to training. NMPNS's licensed operator training program is accredited by the Institute of Nuclear Power Operation (INPO) and is based on a systems approach to training.

The proposed Technical Specifications amendment to re-establish a previously revised commitment to administer the standards of ANSI N18.1–1971 for the position of Manager Operations is also an administrative change. The change does not alter the manner in which the plant systems are operated.

Therefore, the proposed changes do not involve a significant increase in 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 change to clarify the current requirements for licensed operator qualification and the licensed operator training program are administrative changes, and conform to the requirements of 10 CFR [Part] 55. The TS requirements for all other unit staff qualifications remain unchanged.

Although licensed operator qualification and training may have an indirect impact on the possibility of a new or different kind of accident from any accident previously evaluated, the NRC considered this impact during the rule making process, and by promulgation of the revised rule, concluded that this impact remains acceptable as long as the licensed operator training program is accredited and based on a systems approach to training. As previously noted, NMPNS licensed operator training program is accredited by INPO and is based on a systems approach to training.

The proposed TS change to delete a previously approved exception to the qualification requirements contained in ANSI N18.1–1971 for the position of Manager Operations is also an administrative change.

None of the precursors of previously evaluated accidents are affected by these changes, and no new failure modes 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 TS change to update the current requirements applicable to licensed operator qualification and the licensed operator training program are administrative changes. The change is consistent with the requirements of 10 CFR [Part] 55. The TS qualification requirements for all other unit staff remain unchanged.

Licensed operator qualification and training can have an indirect impact on a margin of safety. However, the NRC considered this impact during the rule making process, and by promulgation of the revised 10 CFR [Part] 55, determined that this impact remains acceptable when licensees maintain a licensed operator training program that is accredited and based on a systems approach to training. As previously noted, the NMPNS licensed operator training program is accredited by INPO and is based on a systems approach to training.

The NRC has concluded, as stated in NUREG–1262, ''Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal regulations, Part 55 on Operators' Licenses,'' that the standards and guidelines applied by INPO in their training accreditation program are equivalent to those put forth or endorsed by the NRC. As a result, maintaining an INPO accredited, systems approach based licensed operator training program is equivalent to maintaining an NRC approved licensed operator training program which conforms with applicable NRC Regulatory Guides or NRC endorsed industry standards. The margin of safety is maintained by virtue of maintaining an INPO accredited licensed operator training program.

In addition, the NRC has published NRC Regulatory Issue Summary 2001–01, ''Eligibility of Operator License Applicants,'' dated January 18, 2001, ''to familiarize addressees with the NRC's current guidelines for the qualification and training of reactor operator and senior operator license applicants.'' This document again acknowledges that the INPO National Academy for Nuclear Training (NANT) guidelines for education and experience, outline acceptable methods for implementing the NRC 's regulations in this area.

The proposed Technical Specifications change to re-establish a previously revised plant commitment to administer the standards of ANSI N18.1–1971 for the position of Manager Operations is an administrative change.

The proposed changes do not involve a physical modification of the plant or involve any changes to the methods in which plant systems are operated. The changes do not, in themselves, adversely affect any physical barrier which could contribute to the release of radiation to plant personnel or to the public.

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: Mark G. Kowal.

Nuclear Management Company, LLC, Docket No. 50–282, Prairie Island Nuclear Generating Plant, (PINGP) Unit 1, Goodhue County, Minnesota.

Date of amendment request: August 16, 2007.

Description of amendment request: The proposed amendment would require PINGP monthly Emergency Diesel Generators (EDGS) load test (SR 3.8.1.3) to be performed at or above 90 percent of the diesel generator's continuous power rating. This fulfills the commitment made in the supplement to license amendment request for extension of Technical Specification (TS) 3.8.1, ''AC Sources-Operating,'' Emergency Diesel Generator Completion Time (TAC Nos. MC9001 and MC9002), dated May 10, 2007, Agencywide Documents Access and Management System Accession No. ML071310108.

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.

This license amendment request proposes Technical Specification Surveillance Requirement changes which will increase the monthly test load for the Unit 1 emergency diesel generators to a load greater than 90% of their continuous rated load which is consistent with the guidance of Regulatory Guide 1.9, ''Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants'', Revision 4.

The emergency diesel generators are not accident initiators and therefore, these changes do not involve a significant increase [in] the probability of an accident. The proposed changes increase the test load requirements, are consistent with current regulatory guidance for testing emergency diesel generators, and will continue to assure that this equipment performs its design function. Thus these changes do not involve a significant increase in the consequences of an accident.

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

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

Response: No.

This license amendment request proposes Technical Specification Surveillance Requirement changes which will increase the monthly test load for the Unit 1 emergency diesel generators to a load greater than 90% of their continuous rated load which is consistent with the guidance of Regulatory Guide 1.9, ''Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants'', Revision 4.

The changes proposed for the emergency diesel generators do not change any system operations or maintenance activities. Testing requirements will be revised and will continue to demonstrate that the Limiting Conditions for Operation are met and the system components are functional. The revised test load is consistent with current plant procedures and practices. These changes do not create new failure modes or mechanisms and no new accident precursors are generated.

Therefore, the proposed changes do 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.

This license amendment request proposes Technical Specification Surveillance Requirement changes which will increase the monthly test load for the Unit 1 emergency diesel generators to a load greater than 90% of their continuous rated load which is consistent with the guidance of Regulatory Guide 1.9, ''Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants'', Revision 4.

Current plant procedures require the Unit 1 emergency diesel generators to be load tested above 90% of their continuous rated load each month. This license amendment request proposes to make testing above 90% of the Unit 1 emergency diesel generator's continuous rated load a Technical Specification requirement. Since this change is an increase in the test requirements and the change is consistent with current regulatory guidance, this change does not involve a significant reduction 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 requests involve no significant hazards consideration.

Attorney for licensee: Jonathan Rogoff, Esquire, General Counsel Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Acting Branch Chief: Cliff Munson.

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

Date of amendment request: October 19, 2007, as supplemented by letter dated December 12, 2007.

Description of amendment request: The proposed amendment modifies Technical Specification (TS) 3.6(3), ''Containment Recirculating Air Cooling and Filtering System.'' The licensee has determined that emergency mode (remotely operated) dampers in the containment air cooling and filtering system (CACFS) can be maintained in their accident positions permanently in all plant operating modes. Surveillance Requirement (SR) 3.6.3.a for testing the CACFS emergency mode (remotely operated) dampers each refueling outage will be deleted and be replaced with an SR to verify that the emergency mode dampers are in their accident positions. The licensee also proposes to delete the SR of TS 3.6(3)b to exercise the remotely operated (emergency mode) dampers at intervals not to exceed 3 months.

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 containment air cooling and filtering system (CACFS) is not an initiator of any accident previously evaluated at the Fort Calhoun Station (FCS). The CACFS is an accident mitigation system. The current licensing basis function of the CACFS is to limit the containment pressure rise by providing a means for cooling the containment following a main steam line break (MSLB) design basis accident (DBA).

The CACFS face and bypass dampers will be aligned to their accident positions permanently causing the CACFS to operate in filtered air mode. Surveillance testing has shown that operating the system in this alignment over long periods does not jeopardize filter performance. Over the lifetime of the plant, the differential pressures measured across the combined high efficiency particulate air (HEPA) and charcoal filter banks have met test acceptance criteria.

With the dampers aligned to their accident positions permanently, the removal of TS requirements to check and exercise the dampers does not adversely affect the function of the CACFS. Each refueling outage, the dampers will be verified to be in their accident positions.

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 CACFS was designed to remove heat released to the containment atmosphere during a DBA to the extent necessary to maintain the containment structure below its design pressure. The face and bypass dampers will be aligned in their accident positions permanently, and the air supply, power, and ventilation isolation actuation signal to these dampers will be removed. Thus, the dampers will no longer have an active function and will not be required to change position under accident conditions.

Each refueling outage, the dampers will be verified to be in their accident positions. The CACFS will continue to operate as before except that filter bypass mode will be unavailable. Surveillance testing has shown that the filters are capable of long-term operation in filtered air mode without degrading their ability to respond to a DBA loss-of-coolant accident (LOCA).

No credible new failure mechanisms, malfunctions, or accident initiators not previously considered in the design and licensing basis are created and none of the initial condition assumptions of any accident evaluated in the safety analysis are impacted.

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 containment building and associated penetrations are designed to withstand an internal pressure of 60 psig [pounds per square inch gauge] at 305 °F [degrees Fahrenheit], including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24 hours. The CACFS is credited for maintaining containment pressure and temperatures within design limits. The air coolers are also credited for limiting peak containment pressure for an MSLB.

The CACFS consists of two redundant trains, each train with one air cooling and filtering unit and one air cooling unit, for a total of four cooling units. In accordance with analyses completed for replacement of the FCS steam generators in 2006, operation of the CACFS will continue to be credited in the MSLB containment pressure analysis. The CACFS face and bypass dampers will be aligned to their accident positions permanently. Therefore, TS surveillance requirements to periodically check and exercise these dampers are unnecessary. Each refueling outage, the dampers will be verified to be in their accident positions.

The containment heat removal licensing basis is not adversely affected by the proposed changes. The ability to maintain design limits for containment peak pressure and temperature, as well as long-term containment pressure and temperature, is preserved.

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: James R. Curtiss, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: December 17, 2007.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' and TS 3.6.6, ''Containment Spray and Cooling Systems.'' The Diablo Canyon Power Plant ECCS consists of three separate subsystems: centrifugal charging, safety injection, and residual heat removal. The proposed changes to TS 3.5.2 would add new required actions and extend the Completion Time (CT) of the ECCS from 72 hours to 14 days.

Similarly, the proposed change to TS 3.6.6 involves extending the CT for one inoperable containment spray train from 72 hours to 14 days. These amendments are risk-informed licensing changes.

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 changes increase the Emergency Core Cooling System (ECCS) completion time (CT) to 14 days when one subsystem of one ECCS train is inoperable. Similarly, the proposed changes also increase the containment spray (CS) system CT to 14 days when one CS train is inoperable. These proposed changes do not physically alter any plant structures, systems, or components, and are not accident initiators; therefore, there is no effect on the probability of accidents previously evaluated. When one or more ECCS trains is inoperable, the Technical Specifications (TS) still requires at least 100 percent of the ECCS flow equivalent to a single OPERABLE ECCS train available. Similarly, when one CS train is inoperable, the TS still requires the redundant CS train to be OPERABLE. Therefore, redundant system and subsystems are still able to perform their safety functions. Also the proposed changes do not affect the types or amounts of radionuclides released following an accident, or affect the initiation and duration of their release. Therefore the consequences of accidents previously evaluated, which rely on the ECCS and CS system to mitigate, are not significantly increased.

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 accident from any accident previously evaluated?

Response: No.

There are no new failure modes or mechanisms created due to plant operation with an extended CT. Extended operation with one ECCS train with one subsystem inoperable or with one train of CS system inoperable does not involve any modification to the operational limits or physical design of the systems. There are no new accident precursors generated due to the extended CT.

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 change[s] [are] based upon both a deterministic evaluation and a risk-informed assessment. The deterministic evaluation concluded that though one ECCS train is inoperable for a longer period of time, the availability of the redundant OPERABLE ECCS train can still perform its safety function. Similarly, though one train of the CS system is inoperable for a longer period of time, the redundant OPERABLE CS train can still perform its safety function by providing at least the minimum spray flow to the containment assumed in the accident analyses.

The risk assessment performed to support this license amendment request concluded that the increase in plant risk is small and consistent with the NRC's Safety Goal Policy Statement, ''Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,'' and guidance [contained in] of Regulatory Guides (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.''

Together, the deterministic evaluation and the risk-informed assessment provide assurance that the ECCS and the CS system will still meet their design requirements with the longer CTs proposed.

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

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: December 26, 2007.

Description of amendment request: The proposed amendments would modify the Technical Specification (TS) to establish more effective and appropriate action, surveillance, and administrative requirements related to ensuring the habitability of the control room envelope (CRE) in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' Specifically, the proposed amendments would modify TS 3.7.10, ''Control Room Ventilation System (CRVS),'' and would establish a CRE habitability program in TS Section 5.5, ''Administrative Controls—Programs and Manuals.'' The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process'' associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated December 26, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

Criterion 1—The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The 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 the analysis adopted by the licensee 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 NSHC.

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. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama.

Date of amendment request: November 5, 2007.

Description of amendment request: The proposed amendments would revise Facility Operating License No. NPF–2 and Facility Operating License No. NPF–8 for Farley Nuclear Plant (FNP), Units 1 and 2, specifically, TS Section 5.5.17, ''Containment Leakage Rate Testing Program,'' to resolve a timing conflict between the FNP, Unit 2 R20 refueling outage schedule and the 15-year test date for the FNP, Unit 2 Type A Containment Integrated Leak Rate Test (ILRT), which has a required completion date of March 2010. Although Unit 1 does not have a current timing conflict, a similar Unit 1 change is proposed for consistency.

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 revision to Technical Specifications 5.5.17, ''Containment Leakage Rate Testing Program,'' resolves a schedule conflict between the Farley Nuclear Plant (FNP) Unit 2 refueling outage and the fifteen (15) year Containment Integrated Leak Rate Test date that is currently stated in the FNP Technical Specifications. The previous Integrated Leakage Rate Tests were completed in March 1994 for FNP Unit 1 and March 1995 for FNP Unit 2. A 15 year deferral, granted by Amendments No. 159 and No. 150, placed the next integrated leak rate testing for FNP Unit 1 in March 2009 and FNP Unit 2 in March 2010. Due to minor variations in the refueling outage schedule, the current refueling outage for FNP Unit 2 has been scheduled for April 3, 2010 (Spring 2010). The Type A testing will begin during the FNP Unit 2 refueling outage which is three days after the 15 year time period from the March 1995 date that is currently stated in the revised FNP Technical Specifications (TS). This proposed change will revise FNP TS section 5.5.17 to include the current refueling outage schedule R22 (Spring 2009) for Unit 1 and R20 (Spring 2010) for Unit 2. The proposed Technical Specification 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 reactor 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 reactor containment exists to ensure the plant's ability to mitigate the consequences of an accident, and does not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

Type B and C containment leakage testing will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage 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. FNP test history listed in letter from Southern Nuclear Operating Company to the Nuclear Regulatory Commission dated April 4, 2002 supports this conclusion. The basis and the conclusions reached in the significant hazards evaluation provide in the original SNC amendment request for the ILRT interval extension remain valid and unchanged. Therefore, this change 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.

This proposed change will revise FNP TS section 5.5.17 to include the current refueling outage schedule of R22 for Unit 1 and R20 for Unit 2. The basis and the conclusions reached in the significant hazards evaluation provided in the original amendment request for the ILRT interval extension provided in the original amendment request for the ILRT interval extension remain valid and unchanged.

The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor 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 Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification 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 decrease in a margin of safety?

Response: No.

This proposed change will revise FNP TS section 5.5.17 to include the current refueling outage schedule of R22 for Unit 1 and R20 for Unit 2. The basis and the conclusions reached in the significant hazards evaluation provided in the original amendment request for the ILRT interval extension remain valid and unchanged. The proposed Technical Specifications 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 Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. Type B and C containment leakage testing will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage 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. FNP test history listed in a letter from Southern Nuclear Operating Company dated April 4, 2002 to the Nuclear Regulatory Commission supports this conclusion. Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above, Southern Nuclear Operating Company concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, 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: John Stang, Acting Chief.

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: January 9, 2008.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' Function 7.b, and TS 3.5.4, ''Refueling Water Storage Tank (RWST),'' Surveillance Requirement (SR) 3.5.4.2. The proposed change to TS 3.3.2 lowers the Nominal Trip setpoint and corresponding Allowable Value of the Refueling water Storage Tank (RWST) Level—Low Low at which the semi-automatic switchover from the RWST to the containment emergency sump occurs.

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

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

TS 3.3.2, ''ESFAS Instrumentation,'' Table 3.3.2–1 (page 6 of 7), ''Engineered Safety Feature Actuation System Instrumentation,'' Function 7.b:

No. The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that decreases the Allowable Value and Nominal Trip Setpoint (NTS) of the semi-automatic switchover to containment sump (RWST Level—Low Low) does not have a detrimental impact on the integrity of any plant structure, system, or component (SSC) that initiates an analyzed event. The change does not adversely affect the protective and mitigative capabilities of the plant, nor does the change impact the initiation or probability of occurrence of any accident. The SSCs will continue to perform their intended safety functions.

The minimum containment sump pH used in calculating the radiological consequences for a LOCA remains bounding. The offsite and control room doses will continue to meet the requirements of 10 CFR 100 (Reactor Site Criteria) and 10 CFR 50 Appendix A GDC 19 (General Design Criteria—Control Room).

The proposed AV and NTS for TS Table 3.3.2–1, Function 7.b were determined using an uncertainty methodology previously approved by the NRC for this application. These values provide adequate assurance that required protective and mitigative functions will be initiated as assumed in the transient and accident analyses. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

TS 3.5.4, ''Refueling Water Storage Tank (RWST),'' SR 3.5.4.2:

No. The proposed change that increases the RWST borated water volume does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The RWST borated water volume is not an initiator of any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected.

The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended safety functions to mitigate the consequences of an initiating event within the assumed acceptance limits. The impact on the containment flood level, equipment qualification, and containment sump pH remains within the limits assumed in the design and accident analyses. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

The proposed change will not alter the operation of, or otherwise increase the failure probability of, any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Based on the above discussions, the proposed TS changes do not involve an increase in the consequences of an accident previously evaluated.

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

No. The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. The possibility of a new or different malfunction of safety-related equipment is not created. No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

Based on this evaluation, 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 proposed changes to the semi-automatic switchover to the containment sump RWST Level—Low Low AV and NTS and to the required RWST minimum borated water volume do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside of the design basis. The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intending function to mitigate the consequences of an initiating event within the applicable acceptance criteria.

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. The minimum and maximum pH values remain bounding; therefore, the required amount of trisodium phosphate (TSP) remains unchanged. The impact on the containment flood level, equipment qualification, hydrogen produced by the corrosion of galvanized surfaces and zinc based paints, and chloride induced stress corrosion remains within the limits assumed in the design and accident analyses.

There will be no effect on the manner in which the Safety Limits or Limiting Safety System Settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. 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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: John Stang, Acting Chief.

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.*

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 applications for amendments: February 27, 2007.

Brief description of amendments: These amendments modify Technical Specification (TS) 4.2.1, ''Fuel Assemblies,'' to permit up to four lead fuel assemblies (LFAs) with advanced cladding material to be inserted into the Unit 1 core for operating cycle 19 which is scheduled to begin in April 2008. Two of the LFAs were manufactured by Westinghouse Electric Company and contain a limited number of fuel rods with advanced zirconium-based alloys. The other two LFAs were manufactured by AREVA with fuel rod cladding material classified as M5™ alloy. These LFAs, which were originally inserted into the Unit 2 core in April 2003, remained there for operating cycles 15 and 16 and were subsequently removed in April 2007. These amendments also modify TS 5.6.5, ''Core Operating Limits Report (COLR),'' for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, to include WCAP–15604–NP, ''Limited Scope High Burnup Lead Test Assemblies,'' as an approved analytical method for extended LFA burnup limits.

Date of issuance: December 20, 2007.

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

Amendment Nos.: 283 and 260.

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

Date of initial notice in ***Federal Register:*** *April 24, 2007 (72 FR 20377).*

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated December 20, 2007.

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: February 1, 2007, as supplemented by letter dated August 17, 2007.

Brief description of amendments: These amendments revise Surveillance Requirement 3.5.2.8 in Technical Specification 3.5.2, ''ECCS—Operating,'' to reflect the replacement of the containment recirculation sump suction inlet trash racks and screens with strainers. The containment recirculation sump suction inlet trash racks and screens are being replaced with a strainer design with significantly larger effective surface area in response to Nuclear Regulatory Commission Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.''

Date of issuance: December 27, 2007.

Effective date: As of the date of issuance to be implemented within 60 days following completion of the installation and testing of the plant modifications described in the licensee's letters dated February 1 and August 17, 2007.

Amendment Nos.: 284 and 261.

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

Date of initial notice in Federal Register: March 13, 2007 (72 FR 11385).

The letter dated August 17, 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 these amendments is contained in a Safety Evaluation dated December 27, 2007.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina.

Date of application for amendments: December 21, 2006.

Brief Description of amendments: The amendments change the Technical Specifications (TSs) related to the reactor recirculation system flow balance.

Date of issuance: December 17, 2007.

Effective date: Date of issuance, to be implemented within 60 days.

Amendment Nos.: 244 and 272.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments changed the TSs.

Date of initial notice in ***Federal Register:*** March 13, 2007 (72 FR 11385).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 2007.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York.

Date of application for amendment: July 17, 2007, as supplemented on August 13, 2007.

Brief description of amendment: The proposed amendment would revise action and surveillance requirements related to control room envelope (CRE) habitability in Technical Specification (TS) Section 3.7.3 ''Control Room Emergency Ventilation Air Supply (CREVAS) System,'' and adds a new administrative controls program, TS Section 5.5.14, ''Control Room Envelope Habitability Program.'' In addition, the proposed amendment adds a license condition which specifies the schedule for performing the new surveillance and assessment requirements for the Control Room Envelope Habitability Program, and corrects a typographical error in Appendix C of the license. The changes are consistent with NRC-approved Revision 3 to Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF–448, ''Control Room Habitability.'' TSTF–448, Revision 3 is a proposal to establish more effective and appropriate action, surveillance, and administrative TS requirements related to ensuring the habitability of the control room envelope.

Date of issuance: January 3, 2008.

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

Amendment No.: 289.

Facility Operating License No. DPR–59: The amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 11, 2007 (72 FR 51854).

The August 13, 2007, supplement 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 as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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 application for amendments: January 11, 2007, as supplemented by letters dated August 9, and September 28, 2007.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to support replacement of the steam generators (SGs). They revise TS 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' TS 5.5.9, ''Steam Generator (SG) Program,'' and TS 5.6.10, ''Steam Generator (SG) Tube Inspection Report.''

Date of issuance: January 8, 2008.

Effective date: As of its date of issuance and shall be implemented prior to entry into Mode 4 following the 14th refueling outage for Unit 2 and prior to entry into Mode 4 following the 15th refueling outage for Unit 1.

Amendment Nos.: Unit 1—198; Unit 2—199.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Facility Operating Licenses and Technical Specifications.

Date of initial notice in ***Federal Register:*** February 13, 2007 (72 FR 6787).

The supplemental letters dated August 9, and September 28, 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 January 8, 2008.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: March 6, 2007.

Brief Description of amendments: These amendments authorized revisions to the Updated Final Safety Analysis Report (UFSAR) to permit irradiation of the fuel assemblies beginning with Surry Power Station, Unit Nos. 1 and 2, improved fuel assemblies with ZIRLO (Westinghouse trademark) cladding to a lead rod average burnup of 62,000 MWD/MTU.

Date of issuance: December 19, 2007.

Effective date: As of the date of issuance and shall be implemented within 30 days. The UFSAR changes shall be implemented in the next periodic update made in accordance with 10 CFR 50.71(e).

Amendment Nos.: 257, 256.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments changed the licenses.

Date of initial notice in ***Federal Register:*** March 27, 2007 (72 FR 14309).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 2007.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 51.12(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 application for amendment, (2) the amendment to Facility Operating License, 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's (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*.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. 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, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov*. 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 intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] 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 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.

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

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. Miscellaneous—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

Those permitted to intervene become parties to the proceeding, subject to any

[1] To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 on August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at *HEARINGDOCKET@NRC.GOV*, or by calling (301) 415–1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at *http://www.nrc.gov/site-help/e-submittals/install-viewer.html*. Information about applying for a digital ID certificate is available on NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/apply-certificates.html*.

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at *http://www.nrc.gov/site-help/e-submittals.html*. A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the 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.

Duke Power Company LLC, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1 York County, South Carolina.

Date of amendment request: January 1, 2008, as supplemented January 2, 2008.

Description of amendment request: The amendment approved a one-time extension of the allowed outage time (AOT) for the 1B centrifugal charging (NV) pump beyond the 72 hours allowed by the Technical Specifications (TSs) up to a total of 240 hours as part of the 1B NV pump repair. In addition, the amendment approved a one-time extension for the auxiliary building filtered ventilation exhaust system (ABFVES), to have two ABFVES trains inoperable.

Date of issuance: January 2, 2008.

Effective date: January 2, 2008.

Amendment No.: 239.

Facility Operating License No. (NPF–68): Amendment revised the technical specifications and license.

Public comments requested as to proposed no significant hazards consideration (NSHC): No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated January 2, 2008.

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 Acting Branch Chief: John F. Stang, Acting.

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

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E8–1300 Filed 1–28–08; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Independent External Review Panel to Identify Vulnerabilities in the U.S. Nuclear Regulatory Commission's Materials Licensing Program: Meeting Notice

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Notice of Meeting.

SUMMARY: NRC will convene a meeting of the Independent External Review Panel to Identify Vulnerabilities in the U.S. Nuclear Regulatory Commission's (NRC) Materials Licensing Program on February 8, 2008. A copy of the agenda for the meeting can be obtained by e-mailing Mr. Aaron T. McCraw at the contact information below.

Purpose: To serve as a forum for members of the public to provide oral comments on the Panel's interim observations and recommendations that will be documented in its draft report.

Date and Time for Closed Sessions: February 8, 2008, from 10 a.m. to 12 p.m. This session will be closed so that the Review Panel can receive a classified briefing pursuant to 5 U.S.C. 552b (c)(1).

Date and Time for Open Session: February 8, 2008, from 1:30 p.m. to 3:30 p.m.

Address for Public Meeting: U.S. Nuclear Regulatory Commission, Two White Flint North Building, 11545 Rockville Pike, Rockville, Maryland 20852. Specific room location will be indicated on the agenda.

Public Participation: Any member of the public who wishes to participate in the meeting should contact Mr. McCraw using the information below.

Contact Information: Aaron T. McCraw, e-mail: *atm@nrc.gov,* telephone: (301) 415–1277.

Conduct of the Meeting

Mr. Thomas E. Hill will chair the meeting. Mr. Hill will conduct the meeting in a manner that will facilitate the orderly conduct of business. The following procedures apply to public participation in the meeting:

1. Persons who wish to provide a written statement should submit an electronic copy to Mr. McCraw at the contact information listed above. All submittals must be received by February 1, 2008, and must pertain to the topics on the agenda for the meeting.

2. Questions and comments from members of the public will be permitted during the meeting, at the discretion of the Chairman.

3. The transcript and written comments will be available for inspection at the NRC Public Document Room, 11555 Rockville Pike, Rockville, Maryland 20852–2738, telephone (800) 397–4209, on or about June 1, 2008.

4. Persons who require special services, such as those for the hearing impaired, should notify Mr. McCraw of their planned attendance.

This meeting will be held in accordance with the Atomic Energy Act of 1954, as amended (primarily section 161a); the Federal Advisory Committee Act (5 U.S.C. App); and the Commission's regulations in Title 10, U.S. Code of Federal Regulations, Part 7.

Dated: January 23, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–1499 Filed 1–28–08; 8:45 am]

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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 186th meeting on February 12–14, 2008, at 11545 Rockville Pike, Rockville, Maryland.

Tuesday, February 12, 2008, Room T–2B3

10 a.m.–10:05 a.m.: Opening Remarks by the ACNW&M Chairman (Open)—The Chairman will make opening remarks regarding the conduct of today's sessions.

10:05 a.m.–11:30 a.m.: Semiannual Briefing by the Office of Nuclear Material Safety and Safeguards (NMSS) (Open)—NMSS Office Director and Division Directors will brief the Committee on recent and future activities of interest within their respective programs.

11:30 a.m.–12:00 p.m.: Discussion of ACNW&M Letter Reports (Open)—Discussion of proposed and potential ACNW&M letter reports.

1 p.m.–2:30 p.m.: Draft Guidance on Preventing Legacy Sites (Open)—A representative from the Office of Federal and State Materials and Environmental Management Programs (FSME) will brief the Committee on the draft guidance prepared as part of the Decommissioning Planning and Rulemaking.

2:45 p.m.–4 p.m.: Corrosion of Waste Package and Spent Fuel Dissolution in a Repository Environment (Open)—NRC staff representatives from the Division of High-Level Waste and Repository Safety (DHLWRS), Office of Nuclear Material Safety and Safeguards, will brief the Committee on waste package corrosion and spent fuel dissolution under potential repository conditions.

4 p.m.–5:30 p.m.: Discussion of ACNW&M Letter Reports (Open)—The Committee will discuss potential and proposed ACNW&M letter reports.

Wednesday, February 13, 2008, Room T–2B3

8:30 a.m.–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.–9:30 a.m.: ACNW&M Meeting with NRC Commissioner Peter B. Lyons (Open)—Commissioner Lyons will address the Committee on current topics and issues of common interest.

9:30 a.m.–12 p.m.: Discussion of ACNW&M Letter Reports (Open)—The Committee will discuss potential and proposed ACNW&M letter reports.

1 p.m.–5 p.m.: ACNW&M Working Group Meeting on Managing Low Activity Radioactive Waste (LAW) (Open)—The purpose of this Working Group Meeting is to understand how low-activity radioactive waste (LAW) is being managed in the United States, and to determine if there are ways to improve its management.

1 p.m.–1:15 p.m.: Greetings and Introductions (Open)—Introductory remarks by Dr. Michael Ryan.

1:15 p.m.–2:30 p.m.: Session I: What is LAW (Open)—Dr. Michael Ryan will provide an overview of the expected goals for the Working Group Meeting, the planned technical sessions, and introduce the invited speakers. Two presentations will follow Dr. Ryan's overview.

2:45 p.m.–4:45 p.m.: Session II: Risk-Based Approaches to the Regulation of LAW (Open)—This session includes four presentations.

Thursday, February 14, 2008, Room T–2B3

8:30 a.m.–4:15 p.m.: ACNW&M Working Group Meeting on Managing Low Activity Radioactive Waste (LAW) (Open)—Continued from the previous day.

8:30 a.m.–12 p.m.: Session III: Alternative Disposal Methods for LAW (Open)—Several case studies will be discussed during this session.