designate Antarctic Specially Protected Areas.

The applications received are as follows:

**Permit Application No. 2009–003.**

1. *Applicant:* Sam Feola, Director, Raytheon Polar Services Company, 7400 South Tucson Way, Centennial, CO 80112.

*Activity for Which Permit Is Requested:* Enter Antarctic Specially Protected Areas. The applicant plans to enter Cape Crozier (ASPA 124) to install radio equipment that will provide voice and data services for the science team working in the area. Equipment will be located in the fish hut, as well as a small radio link located approximately 100 yards away on the ridge facing Mt. Terror. Additional visits to the site may be necessary to repair the communications equipment should a failure of the radio links occur.

*Location:* Cape Crozier (ASPA 124).

*Dates:* October 1, 2008 to February 18, 2009.

**Permit Application No. 2009–004.**

2. *Applicant:* Sam Feola, Director, Raytheon Polar Services Company, 7400 South Tucson Way, Centennial, CO 80112.

*Activity for Which Permit Is Requested:* Enter Antarctic Specially Protected Areas. The applicant plans to enter New College Valley, Caughley Beach, Cape Bird (ASPA 116) to install radio equipment that will provide voice and data services for the science team working in the area. Equipment will be located in the fish hut, as well as a small radio link located approximately 75 yards away on the ridge nearest Mt. Bird. Additional visits to the site may be necessary to repair the communications equipment should a failure of the radio links occur.

*Location:* New College Valley, Caughley Beach, Cape Bird (ASPA 116).

*Dates:* October 1, 2008 to February 18, 2009.

**Nadene G. Kennedy,**

*Permit Officer, Office of Polar Programs.*

[FR Doc. E8–9943 Filed 5–5–08; 8:45 am]

**BILLING CODE 7555–01–P**

## NUCLEAR REGULATORY COMMISSION

**[Docket Nos. 50–282 And 50–306]**

### Nuclear Management Company, LLC; Notice of Receipt and Availability of Application for Renewal of Prairie Island Nuclear Generating Plant, Units 1 and 2 Facility Operating Licenses Nos. DPR–42 and DPR–60 for an Additional 20-Year Period

The U.S. Nuclear Regulatory Commission (NRC or Commission) has received an application, dated April 15, 2008, from Nuclear Management Company, LLC, filed pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, and Title 10 of the Code of Federal Regulations Part 54 (10 CFR Part 54), to renew the operating license for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP). Renewal of the licenses would authorize the applicant to operate the facilities for an additional 20-year period beyond the period specified in the current operating licenses. The current operating licenses for PINGP (DPR–42 and DPR–60) expire on August 09, 2013, and October 29, 2014, respectively. PINGP Units 1 and 2 are pressurized-water reactors designed by Westinghouse that are located 28 miles Southeast of Minneapolis, MN. The acceptability of the tendered application for docketing, and other matters including an opportunity to request a hearing, will be the subject of subsequent **Federal Register** notices.

Copies of the application are available to the public at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852 or through the internet from the NRC's Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room under Accession Number ML081050100. The ADAMS Public Electronic Reading Room is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html.* In addition, the application is available at *http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html.* Persons who do not have access to the internet or who encounter problems in accessing the documents located in ADAMS should contact the NRC's PDR Reference staff at 1–800–397–4209, extension 4737, or by e-mail to *pdr@nrc.gov.*

A copy of the license renewal application for the PINGP is also available to local residents near the site at the Red Wing Public Library, 225 East Avenue, Red Wing, MN 55066.

Dated at Rockville, Maryland, this 28th day of April, 2008.

For the Nuclear Regulatory Commission.

**Samson Lee,**

*Acting Director, Division of License Renewal, Office of Nuclear Reactor Regulation.*

[FR Doc. E8–9939 Filed 5–5–08; 8:45 am]

**BILLING CODE 7590–01–P**

## 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 April 10 to April 23, 2008. The last biweekly notice was published on April 22, 2008 (73 FR 21567).

**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 (APS), 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:* January 17, 2008, as supplemented February 29, 2008.

*Description of amendment request:* The proposed amendments would modify the Technical Specifications (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 3.7.11, ''Control Room Essential Filtration System (CREFS),'' and add new TS 5.5.17, ''Control Room Envelope Habitability Program,'' to TS Administrative Controls Section 5.5, ''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 January 17, 2008, 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[s] [Do] Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change[s] [do] not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change[s] [do] 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[s] [revise] the TS for the CRE [essential filtration] 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 [essential filtration] system is the CRE boundary. The CRE [essential filtration] 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 [essential filtration] system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE [essential filtration] 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[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 impact the accident analysis. The proposed change[s] [do] not alter the required mitigation capability of the CRE [essential filtration] 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[s] [do] 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[s] [do] not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, [the] change[s] [do] not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change[s] [Do] Not Involve a Significant Reduction in the Margin of Safety

The proposed change[s] [do] not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change[s] [do] not affect safety analysis acceptance criteria. The proposed change[s] 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[s] [do] 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[s] [do] not involve a significant reduction in a margin of safety. Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a no-significant-hazards consideration.

The NRC staff has reviewed the analysis adopted by the licensee and, based on that review, it appears that the three 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:* 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.

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

*Date of amendment request:* January 22, 2008.

*Description of amendment request:* The proposed amendment would modify the Technical Specification (TS) 3.8.3 requirements related to Diesel Fuel Oil, Lube Oil, and Starting Air by replacing the specific fuel oil and lube oil storage values with the corresponding number of days supply. The specific volumes would be relocated to a licensee-controlled document (*i.e.*, the TS Bases). It would also expand the ''clear and bright'' test in TS 5.5.10 by allowing a water and sediment test to be performed to establish the acceptability of new fuel oil prior to addition to the storage tanks.

*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 to the Diesel Fuel Oil, Lube Oil, and Starting Air Specification relocates the volume of diesel fuel oil and lube oil required to support 7 day operation of the onsite diesel generators, and the volume equivalent to a 6 day supply, to licensee control. The specific volume of fuel oil equivalent to a 7 and 6 day supply is calculated using the NRC approved methodology described in Regulatory Guide 1.137, Revision 1, ''Fuel Oil Systems for Standby Diesel Generators'' and ANSI/ANS [American National Standards Institute/American Nuclear Society] 59.51–1997 (formerly ANSI N195–1976), ''Fuel Oil Systems for Safety-Related Emergency Diesel Generators.'' The specific volume of lube oil equivalent to a 7 and 6 day supply is based on the Emergency Diesel Generator (EDG) manufacturer's consumption values for the run time of the EDG. Because the requirements to maintain a 7 day supply of diesel fuel oil and lube oil are not changed and are consistent with the assumptions in the accident analyses, and the actions taken when the volume of fuel oil and lube oil are less than a 6 day supply have not changed, neither the probability nor the consequences of any accident previously evaluated will be affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the Diesel Fuel Oil Testing Program adds an option to use already approved testing methodology. Since the methodology is already discussed in ASTM D975 [''Standard Specification for Diesel Fuel Oils''] as an acceptable standard to determine water and sediment content, neither the probability nor the consequences of any accident previously evaluated will be affected. 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 changes to the Diesel Fuel Oil, Lube Oil and Starting Air Specification and Diesel Fuel Oil Testing Program do not involve physical alterations of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The changes do not alter assumptions made in the safety analysis but ensure that the diesel generator operates as assumed in the accident analysis. The proposed changes are consistent with the safety analysis assumptions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

*Response:* No.

The proposed change to the Diesel Fuel Oil, Lube Oil, and Starting Air Specification relocates the volume of diesel fuel oil and lube oil required to support 7 day operation of the onsite diesel generators, and the volume equivalent to a 6 day supply, to licensee control. As the bases for the existing limits on diesel fuel oil and lube oil are not changed and the methods used to determine these limits have been previously approved, no change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The proposed change to the Diesel Fuel Oil Testing Program provides an option to use a quantitative method of testing for sediment and water content as an alternative to a qualitative method. This option uses an already accepted method for assessing fuel oil quality. Based on this, there are no alterations to any assumptions used in the accident analysis and this change does not reduce any margin of safety. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

*NRC Branch Chief:* Mark G. Kowal.

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

*Date of amendment request:* February 7, 2008.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TS) Surveillance Requirement (SR) 3.1.3.2 frequency in TS 3.1.3, ''Control Rod OPERABILITY'' from ''7 days after the control rod is withdrawn and THERMAL POWER is greater than the [Low Power Setpoint] LPSP of [Rod Worth Minimizer] RWM'' to ''31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM'' and revise Example 1.4–3 in Section 1.4 ''Frequency'' to clarify the applicability of the 1.25 surveillance test interval extension. The proposed amendment does not adopt the clarification of Source Range Monitor (SRM) TS action for inserting control rods. This clarification was previously adopted during the JAFNPP conversion to Improved Standard Technical Specifications, TS Section 3.3.1.2, required Action E.2, ''Source Range Monitoring [SRM] Instrumentation.''

*Date of publication of individual notice in* **Federal Register**: April 2, 2008 (73 FR 18008).

*Expiration date of individual notice:* May 2, 2008.

*Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas*

*Date of amendment request:* March 13, 2008.

*Description of amendment request:* The licensee proposes to change the Surveillance Requirement (SR) 3.6.5.8 to require verification that the reactor building spray nozzles are unobstructed following maintenance that could result in nozzle blockage in lieu of the current SR of performing the test every 10 years.

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

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

*Response:* No.

The Reactor Building Spray System is not an initiator of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plan structure, system, or component that may initiate an analyzed event. The proposed change will not alter the operation or otherwise increase the failure probability of any plant equipment that can initiate an analyzed accident. This change does not affect the plant design. There is no increase in the likelihood of formation of significant corrosion products. Due to their location at the top of the containment, introduction of foreign material into the spray headers is unlikely. Foreign materials exclusion controls during and following maintenance provides assurance that the nozzles remain unobstructed. Consequently, there is no significant increase in the probability of an accident previously evaluated.

The Reactor Building Spray system is designed to address the consequences of a Loss of Coolant Accident (LOCA) or a Main Steamline Break (MSLB) inside the reactor building. The Reactor Building Spray system is capable of performing its function effectively with the single failure of any active component in the system, any of its subsystems, or any of its support systems.

Therefore, the consequences of an accident previously evaluated are not significantly affected by the proposed change.

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 will not physically alter the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation.

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

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

*Response:* No.

The system piping and nozzles are made if material that is not susceptible to corrosion. Obstruction from sources external to the system is highly unlikely due to the location high in the reactor building and not being readily accessible. Strict controls are established to ensure the foreign material is not introduced into the Reactor Building Spray system during maintenance or repairs. Maintenance activities that could introduce significant foreign material into the system require subsequent system cleanliness verification which would prevent nozzle blockage. The spray header nozzles are expected to remain unblocked and available in the event that the safety function is required. The capacity of the system would remain unaffected.

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

*Attorney for licensee:* Terence A. Burke, Associate General Counsel—Entergy Nuclear Operations, P.O. Box 31995, Jackson, Mississippi 39286–1995.

*NRC Branch Chief:* Thomas G. Hiltz.

*Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas*

*Date of amendment request:* March 13, 2008.

*Description of amendment request:* The proposed changes would replace the current Technical Specification (TS) 3.4.12, ''RCS [Reactor Coolant System] Specific Activity'' limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xe–133 (DEX) definition that would replace the current E Bar average disintegration energy definition. In addition, the current dose equivalent I–131 (DEI) definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs).

*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.

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

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 in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for a new or different kind of accident from any previously calculated.

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 limits on noble gas radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses.

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

*Attorney for licensee:* Terence A. Burke, Associate General Counsel—Entergy Nuclear Operations, P.O. Box 31995, Jackson, Mississippi 39286–1995.

*NRC Branch Chief:* Thomas G. Hiltz.

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

*Date of amendment request:* March 13, 2008.

*Description of amendment request:* The proposed changes would replace the current TS 3.4.8, ''Reactor Coolant System Specific Activity'' limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xe-133 (DEX) definition that would replace the current E Bar average disintegration energy definition. In addition, the current dose equivalent I–131 (DEI) definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs).

*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.

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

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 in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter. The change does not create the potential for a new or different kind of accident from any previously calculated.

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 limits on noble gas radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses.

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

*Attorney for licensee:* Terence A. Burke, Associate General Counsel—Entergy Nuclear Operations, P. O. Box 31995, Jackson, Mississippi 39286–1995.

*NRC Branch Chief:* Thomas G. Hiltz.

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

*Date of amendment request:* March 13, 2008.

*Description of amendment request:* The proposed change will relocate Technical Specification (TS) 3.4.7, ''Reactor Coolant System Chemistry,'' to the Technical Requirements Manual (TRM).

*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 acts to relocate current Reactor Coolant System (RCS) chemistry limits and monitoring requirements from the TSs to the TRM. Monitoring and maintaining RCS chemistry minimizes the potential for corrosion of RCS piping and components. Corrosion effects are considered a long-term impact on RCS structural integrity. Because RCS chemistry will continue to be monitored and controlled, relocating the current TS requirements to the TRM will not present an adverse impact to the RCS and, subsequently, will not impact the probability or consequences of an accident previously evaluated. Furthermore, once relocated to the TRM, changes to RCS chemistry limits or monitoring requirements will be controlled in accordance with 10 CFR 50.59.

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 result in any plant modifications or changes in the way the plant is operated. The proposed change only acts to relocate current RCS chemistry limits and monitoring requirements from the TSs to the TRM.

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

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

*Response:* No.

The proposed change will maintain limits on RCS chemistry parameters and will continue to provide associated monitoring requirements. Once relocated to the TRM, changes to RCS chemistry limits or monitoring requirements will be controlled in accordance with 10 CFR 50.59. In addition, the RCS chemistry limits are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

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

*Attorney for licensee:* Terence A. Burke, Associate General Counsel—Entergy Nuclear Operations, P.O. Box 31995, Jackson, Mississippi 39286–1995.

*NRC Branch Chief:* Thomas G. Hiltz.

*Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania*

*Date of amendment request:* December 12, 2007.

*Description of amendment request:* The proposed changes are administrative in nature and provide editorial changes to the technical specifications (TSs). The proposed changes involve: (1) Correcting the index; (2) removing cycle specific requirements or notes that have since expired and are no longer applicable; (3) deleting references to previously deleted requirements; (4) changing references to the location of previously relocated information; and (5) other editorial corrections. These proposed changes correct minor inconsistencies that have been introduced over time as a result of previous changes to the TSs or involve changes that are solely editorial in nature.

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

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

*Response:* No.

The proposed changes are administrative in nature and do not impact the physical configuration or function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, maintained, modified, tested, or inspected. The proposed changes do not impact the initiators or assumptions of analyzed events, nor do they impact mitigation of accidents or transient events.

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

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

*Response:* No.

The proposed changes are administrative in nature and do not alter plant configuration, require that new plant equipment be installed, alter assumptions made about accidents previously evaluated, or impact the function of plant SSCs or the manner in which SSCs are operated, maintained, modified, tested, or inspected. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

*Response:* No.

The proposed changes are administrative in nature and do not involve any physical changes to plant SSCs or the manner in which SSCs are operated, maintained, modified, tested, or inspected. The proposed changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions of operation, or design parameters for any SSC. The proposed changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting an accident analysis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

*NRC Branch Chief:* Harold K. Chernoff.

*Exelon Generation Company, LLC, and PSEG Nuclear, LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station (PBAPS),*

*Units 2 and 3, York and Lancaster Counties, Pennsylvania*

*Date of amendment request:* July 13, 2007.

*Description of amendment request:* The proposed amendment would modify the Technical Specifications to support application of Alternative Source Term (AST) methodology at PBAPS Units 2 and 3. The fission product release from the reactor core into containment is referred to as the ''source term,'' and is characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release from the reactor core as discussed in Technical Information Document (TID) 14844, ''Calculation of Distance Factors for Power and Test Reactor Sites.'' Since the publication of TID 14844, advances have been made in understanding the composition and magnitude, chemical form, and timing of fission product releases from severe nuclear power plant accidents. In light of these insights, NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' was published in 1995 with revised ASTs for use in the licensing of future light-water reactors.

The Nuclear Regulatory Commission (NRC), in Title 10 of the Code of Federal Regulations, Section 50.67 (10 CFR 50.67), ''Accident source term,'' subsequently allowed the use of the ASTs described in NUREG–1465 at operating plants. This request to apply the AST methodology is made in accordance with 10 CFR 50.67, with the exception that TID 14844 will continue to be used as the radiation dose basis for equipment qualification at PBAPS Units 2 and 3. Application of the AST methodology at PBAPS Units 2 and 3 requires that radiation dose limits specified in 10 CFR 50.67 are adhered to for the exclusion area boundary, the low population zone outer boundary, and the facility control room.

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

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

The implementation of alternative source term (AST) assumptions has been evaluated in revisions to the analyses of the following limiting design basis accidents (DBAs) at Peach Bottom Atomic Power Station (PBAPS):

• Loss-of-Coolant Accident,
• Fuel Handling Accident,
• Control Rod Drop Accident, and
• Main Steam Line Break Accident.

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1. The Alternative Source Term is an input to calculations used to evaluate the consequences of an accident, and does not by itself affect the plant response, or the actual pathway of the radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Therefore, the consequences of an accident previously evaluated are not significantly increased.

The equipment affected by the proposed changes is mitigative in nature, and relied upon after an accident has been initiated. Application of the Alternative Source Term (AST) does not involve any physical changes to the plant design. While the operation of various systems do change as a result of these proposed changes, these systems are not accident initiators. Application of the AST is not an initiator of a design basis accident. The proposed changes to the Technical Specifications (TS), while they revise certain performance requirements, do not involve any physical modifications to the plant. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. As such, removal of operability requirements during the specified conditions will not significantly increase the probability of occurrence for an accident previously analyzed. Since design basis accident initiators are not being altered by adoption of the Alternative Source Term analyses, the probability of an accident previously evaluated is not affected.

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

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes). Similarly, it does not physically change any structures, systems or components involved in the mitigation of any accidents; thus, no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

As such, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences due to design basis accidents comply with the requirements of 10 CFR 50.67 and the guidance of Regulatory Guide 1.183. The proposed amendment is associated with the implementation of a new licensing basis for PBAPS Design Basis Accidents (DBAs). Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, ''Accident Source Term'', and Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area boundary (EAB) and low population zone boundary (LPZ), as well as the Control Room, are within corresponding regulatory limits.

Therefore, operation of PBAPS in accordance with the proposed changes will 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. J. Bradley Fewell, Associate General Counsel, Exelon Generation Company LLC, 4300 Winfield Road, Warrenville, IL 60555.

*NRC Branch Chief:* Harold K. Chernoff.

*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:* March 31, 2008.

*Description of amendment request:* FPL Energy Point Beach, LLC, requests adoption of an approved change to the Standard Technical Specifications (STS) for pressurized-water reactor (PWR) plants (NUREG–1430, NUREG–1431, & NUREG–1432) and plant-specific technical specifications (TS), to replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on dose equivalent Xenon-133 and would take into account only the noble gas activity in the primary coolant. In addition, the current dose equivalent I–131 definition would be revised to allow the use of additional thyroid dose conversion factors. The changes are consistent with Nuclear Regulatory Commission (NRC)-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–490, Revision 0.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated Reactor coolant specific activity is not an initiator for any accident previously evaluated. The Completion Time when primary coolant gross activity is not within limit is not an initiator for any accident previously evaluated. The current variable limit on primary coolant iodine concentration is not an initiator to any accident previously evaluated. As a result, the proposed change does not significantly increase the probability of an accident. The proposed change will limit primary coolant noble gases to concentrations consistent with the accident analyses. The proposed change to the Completion Time has no impact on the consequences of any design basis accident since the consequences of an accident during the extended Completion Time are the same as the consequences of an accident during the Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

The proposed change in specific activity limits does not alter any physical part of the plant nor does it affect any plant operating parameter.

The change does not create the potential for a new or different kind of accident from any previously calculated.

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

The proposed change revises the limits on noble gas radioactivity in the primary coolant. The proposed change is consistent with the assumptions in the safety analyses and will ensure the monitored values protect the initial assumptions in the safety analyses. Based upon the reasoning presented above, the requested change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the 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 Branch Chief:* Lois M. James.

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

*Date of amendment request:* March 31, 2008.

*Description of amendment request:* The licensee proposed to increase the current maximum power level authorized by Section 2.C(1) of the renewed facility operating license from 1,775 megawatts thermal (Mwt) to 1,870 Mwt, an approximately five percent increase from the current licensed thermal power. The current maximum power level of 1,775 Mwt was approved in 1998, an increase of 6.3 percent from the original licensed thermal power of 1670 Mwt. Thus, when approved, the licensee's proposed amendment would take the maximum power level to about 12 percent above the original license thermal power. The licensee's application addresses in details each of the following major technical areas: Extended power uprate, containment analysis methods change, increase in credit for containment overpressure for low head emergency core cooling system (ECCS) pumps, and reactor internal pressure differentials (RIPDs) for the steam dryer.

*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). The licensee's NSHC analysis, addressing each technical area listed above, is reproduced below:

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

*Extended Power Uprate*

*Response:* No.

The probability (frequency of occurrence) of [d]esign [b]asis [a]ccidents occurring is not affected by the increased power level, because Monticello Nuclear Generating Plant (MNGP) continues to comply with the regulatory and design basis criteria established for plant equipment. A probabilistic risk assessment demonstrates that the calculated core damage frequencies do not significantly change due to [e]xtended [p]ower [u]prate (EPU). Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to EPU. No new challenges to safety-related equipment result from EPU.

The changes in consequences of postulated accidents, which would occur from 102 percent of the EPU [rated thermal power] RTP compared to those previously evaluated, are acceptable. The results of EPU accident evaluations do not exceed the NRC[-] approved acceptance limits. The spectrum of postulated accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of fuel and core design, for example, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDL) are still met. Continued compliance with the SLMCPR and other SAFDLs will be confirmed on a cycle[-]specific basis consistent with the criteria accepted by the NRC.

Challenges to the [r]eactor [c]oolant [p]ressure [b]oundary were evaluated at EPU conditions (pressure, temperature, flow, and radiation) and were found to meet their acceptance criteria for allowable stresses and overpressure margin. Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet the current licensing basis. The increase in the calculated post[-] LOCA suppression pool temperature above the currently assumed peak temperature was evaluated and determined to be acceptable. Radiological release events (accidents) have been evaluated, and have been shown to meet the guidelines of 10 CFR 50.67.

*Containment Analysis Methods Change*

*Response:* No.

The use of passive heat sinks, variable RHR [residual heat removal] heat exchanger capability K-value, and mechanistic heat and mass transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long[-]term design[-] basis [-]accident loss of coolant accident (DBA–LOCA) containment analysis are not relevant to accident initiation, but rather, pertain to the method used to accurately evaluate postulated accidents. The use of these elements does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of the containment response to DBA–LOCAs. Therefore, the containment analysis method change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

*Increase in Credit for Containment Overpressure for Low Head Emergency Core Cooling System (ECCS) Pumps*

*Response:* No.

These changes update parameters used in the MNGP safety analyses and expand the range and scope of the analyses. This will result in a more realistic analysis of available containment overpressure under design [-]basis accident conditions. The updated analyses affect only the evaluation of previously reviewed accidents. No plant structure, system, or component (SSC) is physically affected by the updated and expanded analyses. No method of operation of any plant SSC is affected. Therefore, there is no significant increase in the probability or consequence of a previously evaluated accident.

*Reactor Internal Pressure Differentials (RIPDs) for the Steam Dryer*

*Response:* No.

The revised steam dryer RIPDs are used in evaluating loads in reactor vessel internals for various conditions (*i.e.*, during normal, upset and faulted conditions). The values more accurately represent the actual plant configuration. No plant structure, system, or component (SSC) is physically affected by the updated and expanded analyses. No method of operation of any plant SSC is affected. Therefore, there is no significant increase in the probability or consequence of a previously evaluated accident.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 Mwt, which bounds this license amendment request to operate at 1,870 Mwt. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

*Extended Power Uprate*

*Response:* No.

Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. EPU uses developed technology and applies it within capabilities of existing or modified plant safety[-]related equipment in accordance with the regulatory criteria (including NRC[-]approved codes, standards and methods). No new accidents or event precursors have been identified.

The MNGP TS require revision to implement EPU. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

*Containment Analysis Methods Change*

*Response:* No.

The use of passive heat sinks, variable RHR heat exchanger capability K-value, and mechanistic heat and transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long term DBA–LOCA containment analysis are not relevant to accident initiation, but pertain to the method used to evaluate currently postulated accidents. The use of these analytical tools does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated. Further, they do not result in the need to postulate any new accident scenarios. Therefore, the containment analysis method change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

*Increase in Credit for Containment Overpressure for Low Head ECCS Pumps*

*Response:* No.

The proposed change involves the updating and expansion in scope of the existing design bases analysis with respect to the available containment overpressure. No new failure mode or mechanisms have been created for any plant SSC important to safety nor has any new limiting single failure been identified as a result of the proposed analytical changes. Therefore, the change to containment overpressure credited for low pressure ECCS pumps does not create the possibility of a new or different kind of accident from any accident previously evaluated.

*Reactor Internal Pressure Differentials for the Steam Dryer*

*Response:* No.

The revised steam dryer RIPDs are used in evaluating loads in reactor vessel internals for various conditions (i.e., during normal, upset and faulted conditions). The steam dryer RIPDs are not relevant to accident initiation, but only pertain to the method used to evaluate reactor vessel internals loads. The revised steam dryer RIPD values more accurately represent the actual plant configuration. Therefore, the change to steam dryer RIPDs does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 Mwt, which bounds this license amendment request to operate at 1,870 Mwt. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

*Extended Power Uprate*

*Response:* No.

The EPU affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for EPU conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for design[-]basis event categories. No NRC acceptance criterion is exceeded. Because the MNGP configuration and responses to transients and postulated accidents do not result in exceeding the presently approved NRC acceptance limits, the proposed changes do not involve a significant reduction in a margin of safety.

*Containment Analysis Methods Change*

*Response:* No.

The use of passive heat sinks, variable RHR heat exchanger capability K-value, and mechanistic heat and mass transfer from the suppression pool surface to the wetwell airspace after 30 seconds for the long[-]term DBA–LOCA containment analysis are realistic phenomena and provide a conservative prediction of the plant response to DBA–LOCAs. The increase in pressure and temperature are relatively small and are within design limits. Therefore, the containment analysis methods change does not involve a significant reduction in the margin of safety.

*Increase in Credit for Containment Overpressure for Low Head ECCS Pumps*

*Response:* No.

The proposed changes revise containment response analytical methods and scope for containment pressure to assist in ECCS pump net positive suction head (NPSH). The changes are still based on conservative but more realistic analysis of available containment overpressure determined using analysis methods that minimize containment pressure and maximize suppression pool temperature. These changes do not constitute a significant reduction in the margin of safety.

*Reactor Internal Pressure Differentials for the Steam Dryer*

*Response:* No.

The revised steam dryer RIPDs are used in evaluating loads in reactor vessel internals for various conditions (i.e., during normal, upset and faulted conditions). The revised steam dryer RIPD values more accurately represent the actual plant configuration. The changes are still conservative but more accurately represent the MNGP configuration. These changes do not constitute a significant reduction in the margin of safety.

The analyses supporting the above evaluations were performed at the EPU power level of 2,004 Mwt, which bounds this license amendment request to operate at 1,870 Mwt. 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 the NRC staff's own analysis above, 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:* Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401.

*NRC Branch Chief:* Lois M. James.

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

*Date of amendment request:* April 3, 2008.

*Description of amendment request:* The proposed amendment would modify Technical Specifications (TS) requirements related to control room envelope (CRE) habitability in TS Section 3.7.4, ''Control Room Emergency Filtration (CREF) 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:* Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401.

*NRC Branch Chief:* Lois M. James.

*PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania*

*Date of amendment request:* March 28, 2008.

*Description of amendment request:* The amendments would revise PPL Susquehanna, LLC, Units 1 and 2 (PPL) Technical Specifications (TSs) 3.8.4, ''DC Sources—Operating,'' to establish two new Conditions, A and B the associated Required Actions with their completion times, and also, make some editorial and administrative 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes revise the Technical Specifications (TS) for the DC Electrical Power Systems and propose new Actions with increased completion times for an inoperable battery charger. The DC electrical power systems, including associated battery chargers, are not initiators to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). Operation in accordance with the proposed TS ensures that the DC electrical power systems are capable of performing functions as described in the FSAR. Therefore, the mitigative functions supported by the DC Power Systems will continue to provide the protection assumed by the analysis. The integrity of fission product barriers, plant configuration, and operating procedures as described in the FSAR will not be affected by the proposed changes.

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

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

No. The proposed changes only involve revising the TS for the DC electrical power systems. The DC electrical power systems are used to supply equipment used to mitigate an accident. These mitigative functions, supported by the DC electrical power systems are not affected by these changes and they will continue to provide the protection assumed by the safety analysis described in the FSAR. There are no new types of failures or new or different kinds of accidents or transients that could be created by these changes.

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

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

No. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC electrical system capacity is ensured to support operation of mitigation equipment. The equipment fed by the DC electrical sources will continue to provide adequate power to safety related loads in accordance with the safety analysis.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, 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:* Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

*NRC Branch Chief:* Mark G. Kowal.

*PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania*

*Date of amendment request:* March 28, 2008.

*Description of amendment request:* The amendments would revise PPL Susquehanna, LLC, Units 1 and 2 (PPL) Technical Specifications (TSs) TS 3.6.4.1 ''Secondary Containment,'' and TS 3.6.4.3 ''Standby Gas Treatment System,'' as follows:

(1) To add a new Required Action option for TS 3.6.4.1 Condition A, to allow additional time to restore secondary containment to OPERABLE when the inoperability is not caused by a loss of secondary containment integrity,

(2) To add a new Actions note TS 3.6.4.1, to allow opening of secondary containment heating ventilation and air conditioning duct access doors and opening of a secondary containment equipment ingress/egress door (102 door) under administrative controls provided no movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, or operations with a potential for draining the reactor vessel (OPDRVs) are in progress,

(3) To modify the existing note to Surveillance Requirement (SR) 3.6.4.1.3 and add a second note to this same SR, to expand upon the existing SR exception note by adding other types of door access openings that occur for entry and exit of people or equipment, and

(4) The administrative change to remove a one-time allowance in TS 3.6.4.1 and TS 3.6.4.3 ''Standby Gas Treatment System [SGTS],'' that extended the allowable Completion Time for Secondary Containment

inoperable and two SGTS subsystems inoperable in MODE 1, 2, or 3. This allowance was previously incorporated into both Unit 1 and Unit 2 TSs to facilitate Reactor Recirculating Fan Damper Motor work.

*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.

These changes do not involve any physical change to structures, systems, or components (SSCs) and do not alter the method of operation of any SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No SSC failure modes or mechanisms are being introduced, and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed new Required Action option for TS 3.6.4.1 Condition A and the Notes that are being modified and added in both the Unit 1 and Unit 2 Technical Specifications ensures that the secondary containment remains capable of performing its function. The Required Action change, which will permit up to 72 hours to restore secondary containment vacuum, only provides this additional time when it can be shown that the vacuum loss has not been caused through compromise of the secondary containment boundary.

The proposed Note modifications and additions addressing secondary containment access door and duct access door openings will provide relief from TS requirements that must currently be implemented in response to various routine plant activities. These activities can be managed through administrative controls that will ensure doors can be closed quickly (within 30 minutes) to re-establish secondary containment before the early in-vessel release phase begins (Regulatory Guide 1.183).

These changes do not, therefore, result in an increase in the probability or consequences of an accident previously evaluated.

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

*Response:* No.

The proposed changes do not involve a physical alteration of any plant equipment. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change does not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR [final safety analysis report]. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis.

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

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

*Response:* No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes are acceptable because the Completion Time for the new Required Action to verify secondary containment boundary integrity within 4 hours has been established to be consistent with the current completion time of Condition A. A failure or inability to complete this verification will result in the implementation of LCO [limiting condition for operation] 3.6.4.1 requirements in the same timeframe that currently exists. Upon successful completion of this verification, however, the proposed change will provide 72 hours to restore secondary containment to an operable status through vacuum restoration. When in this condition, the secondary containment and SGTS are capable of performing their design basis function.

The Note modifications and additions to TS 3.6.4.1 are also acceptable because the revised Notes provide allowances and exemptions to Technical Specification entry for routine plant activities that can be administratively controlled and quickly restored.

The plant response to analyzed events is not affected by these changes and will, continue to provide the margin of safety assumed by the safety analysis.

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:* Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

*NRC Branch Chief:* Mark G. Kowal.

*Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California*

*Date of amendment request:* November 30, 2007.

*Description of amendment request:* The proposed Technical Specification changes will provide operational flexibility supported by direct current (DC) electrical subsystem design upgrades that are in progress. These upgrades will provide increased capacity batteries, additional battery chargers, and the means to cross-connect DC subsystems while meeting all design battery loading requirements. With these modifications in place, it will be feasible to perform routine surveillances as well as battery replacements online.

*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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

*Response:* No.

The proposed changes to Technical Specifications (TS) 3.8.4 and 3.8.6 would allow extension of the Completion Time (CT) for inoperable Direct Current (DC) distribution subsystems to manually cross-connect DC distribution buses of the same safety train of the operating unit for 21 days (30 days for upgrade to 1800 amp-hour rated batteries). Currently the CT only allows for 2 hours to ascertain the source of the problem before a controlled shutdown is initiated. Loss of a DC subsystem is not an initiator of an event. However, complete loss of a Train A (subsystems A and C) or Train B (subsystems B and D) DC system would initiate a plant transient/plant trip.

Operation of a DC Train in cross-connected configuration does not affect the quality of DC control and motive power to any system. Therefore, allowing the cross-connect of DC distribution systems does not significantly increase the probability of an accident previously evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR).

The above conclusion is supported by Probabilistic Risk Assessment (PRA) evaluation which encompasses all accidents, including UFSAR Chapter 15.

New TS Surveillance Requirement (SR) 3.8.4.4 is added to allow the application of the modified performance discharge testing on batteries rated at 1800 amp-hour using a frequency of 30 months. The application of the modified performance test is the preferred choice at SONGS for Class 1 E 1800 amp-hour rated batteries. Therefore, only the modified performance discharge test will be used which uses the combined duty cycle of the cross-connected subsystems AC or B–D. Battery life expectancy is optimized by using a 30-month modified performance test (service and performance test combined). The more rigorous modified performance discharge test will be applied in intervals of 30 months over the entire battery life. Using the same test method and test frequency throughout the battery life ensures that best trending results are achieved. The test frequency of 30 months will better correspond with scheduling of the more rigorous 60-month interval battery performance of modified performance discharge tests. Based on operating experience, the interval of 30 months is not expected to affect SONGS' capability to detect battery health and capacity.

The relocation of preventive maintenance surveillances and certain operating limits and actions to the Licensee Controlled Specifications and new Battery Monitoring and Maintenance Program will not challenge the ability of the DC electrical power system to perform its design function. Appropriate monitoring and maintenance consistent with industry standards will continue to be performed. In addition, the DC electrical power system is within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure the control of maintenance activities associated with the DC electrical power system. Enhancements from TSTF–360, Rev. 1 and IEEE 450–2002 have been incorporated into TSs 3.8.4, 3.8.5, and 3.8.6. These changes do not impact the probability or consequences of an accident previously evaluated.

Further, changes are made of an editorial nature or provide clarification regarding electrical 'Trains' and 'Subsystems' by using a more conventional terminology. TSs affected by editorial changes include 3.8.1, 3.8.4, 3.8.5, 3.8.6, 3.8.7, 3.8.9, and 3.8.10. The changes being proposed in the TS do not affect assumptions contained in other safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Will operation of the facility in accordance with this proposed change create the possibility of new or different kind of accident from any accident previously evaluated?

*Response:* No.

The proposed changes involve restructuring the TS for the DC electrical power system. The DC electrical power system, including associated battery chargers, is not an initiator to any accident sequence analyzed in the UFSAR. Rather, the DC electrical power system is used to supply equipment used to mitigate an accident.

The proposed change modifies TSs and surveillances for batteries and chargers to meet the improvements of TSTF–360, Rev. 1 and IEEE 450–2002 whose intent is to maintain the same equipment capability as previously assumed in Southern California Edison's (SCE's) commitment to IEEE 450–1980.

The proposed change will allow the cross-tie of DC subsystems and allow extension of the CT for an inoperable subsystem to 21 days (30 days for upgrade to 1800 amp-hour rated batteries). Failure of the cross-tied DC buses and/or associated battery(ies) is bounded by existing evaluations for the failure of an entire electrical train.

Swing battery chargers are added to increase the overall DC system reliability. Administrative and mechanical controls are in place to ensure the design and operation of the DC systems continue to meet the UFSAR design basis.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

*Response:* No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The changes associated with the new battery maintenance and monitoring program will ensure that the station batteries are maintained in a highly reliable manner. The equipment fed by the DC electrical sources will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

Improvements in accordance with IEEE 450–2002 and TSTF–360, Rev. 1 maintain the same level of equipment performance stated in the UFSAR and the current Technical Specifications.

The addition of swing battery chargers increases the overall DC system reliability. Administrative and mechanical controls will be in place to ensure that the design and operation of the DC systems continue to meet the UFSAR design basis.

The addition of the DC cross-tie capability proposed for TS 3.8.4 has been evaluated, as described previously, using PRA and determined to be of acceptable risk as long as the duration while cross-tied is limited to 30 days. A new Condition has been included as part of this proposed change to ensure that plant operation, with DC buses cross-tied, will not exceed 21 days (30 days for upgrade to 1800 amp-hour rated batteries).

Revising the LCO statement to reflect the SONGS-specific design terminology and renaming existing conditions to make the Condition more consistent with the Standard Technical Specifications (STS) is considered administrative.

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

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

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

*NRC Branch Chief:* Thomas G. Hiltz.

*Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia*

*Date of amendment request:* August 29, 2006, as supplemented November 6, November 27, 2006, January 30, June 22, July 16, August 13, October 18, December 11, 2007, January 24, February 4, February 25 (two letters, nos. 1389 and 0175), February 27, and March 13, 2008.

*Description of amendment request:* The proposed amendments would revise the licensing and design basis, including the Technical Specifications, with a full scope implementation of an alternative source term (AST). The licensee states that the AST analyses include determination of the onsite radiological doses, specifically the main control room, technical support center and off-site radiological doses resulting from the loss-of-coolant, main steam line break, control rod drop, and fuel-handling design-basis accident (DBA) analyses. The licensee states that the analyses demonstrate that, using AST methodologies, the post-accident onsite and offsite doses remain within regulatory acceptance limits.

*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?

Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs. The AST does not affect the design or manner in which the facility is operated; rather, once the occurrence of an accident has been postulated, the new accident source term is an input to analyses that evaluate the radiological consequences. The implementation of the AST and changed Technical Specifications have been incorporated in the analyses for the limiting DBAs at HNP. The structures, systems, and components affected by the proposed change are mitigative in nature and relied upon after an accident has been initiated. Based on the revised analyses, the proposed changes to the Technical Specifications (including revised leakage limits) impose certain performance criteria which do not increase accident initiation probability. The proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter 15 of the Unit 2 Final Safety Analysis Report. Therefore, the proposed change does not result in an increase in the probability of an accident previously identified. Plant specific AST radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the regulatory guidance provided by the Nuclear Regulatory Commission for use with the AST. This guidance is presented in [Title 10 of the Code of Federal Regulations, Section 50.67] (10 CFR 50.67), [Accident Source Term] Regulatory Guide 1.183, [Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (ML003716792)] and Standard Review Plan, Section 15.0.1. Therefore, the proposed change does not result in 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 previously evaluated?

Implementation of AST and associated changes does not alter or involve any design basis accident initiators. These changes do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed 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 decrease in the margin of safety?

The changes proposed are associated with a revision to the licensing basis for HNP. Approval of the licensing basis change from the original source term to the AST is requested by this application for a license amendment. The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies and conservative inputs to ensure that analyzed events are bounding and safety margin has been retained. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan 15.0.1. Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in the margin of safety.

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

*Attorney for licensee:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Branch Chief:* Melanie C. Wong.

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

*Date of amendment request:* March 27, 2008.

*Description of amendment request:* The proposed amendment would revise the allowable value for Function 3, ''Containment Purge Exhaust Radiation Monitors,'' in Technical Specifications (TSs) Table 3.3.6–1, ''Containment Vent Isolation Instrumentation,'' of Limiting Conditions for Operation 3.3.6, during Modes 1 through 4. The current allowable value was found to be non-conservative for operating Modes 1 through 4 because the basis for the specified value inappropriately credited the containment purge exhaust filters, which are only required during movement of irradiated fuel assemblies within containment. The current allowable value remains acceptable during movement of irradiated fuel assemblies within containment.

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

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

*Response:* No.

The proposed change is associated with radiation effluent monitoring and isolation of Containment Purge exhaust flow in the event of a design basis SBLOCA [small break loss of coolant accident]. The change is not associated with equipment or processes which can initiate a design basis accident. Consequently, this change does not affect the probability of an accident previously evaluated.

The revised purge exhaust monitor allowable value will ensure the monitors isolate the purge exhaust and will limit the offsite doses associated with a SBLOCA to well within the limits of 10 CFR 100. This change serves to ensure the consequences of an accident previously evaluated remain bounded by the plant's current licensing basis. Therefore, the consequences of accidents previously evaluated are not increased by this change.

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

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

*Response:* No.

The proposed change is associated with radiation effluent monitoring and isolation of Containment Purge exhaust flow in the event of a design basis SBLOCA. The change is not associated with equipment or processes which can initiate a design basis accident. The change does not introduce new accident initiators or physical changes in plant equipment.

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

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

*Response:* No.

The proposed change involves a conservative change in the Containment Purge exhaust radiation monitor allowable value in TS Table 3.3.6–1. The new allowable value reflects a change in the monitor analytical limit which does not assume credit for the Containment Purge exhaust filters. The proposed allowable value will ensure the monitors will isolate the purge exhaust as assumed in the existing design basis SBLOCA analysis.

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:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

*NRC Branch Chief:* L. Raghavan.

## 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.*

*AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania*

*Date of application for amendment:* April 12, 2007.

*Brief description of amendment:* The amendment modifies the TMI–1 technical specifications related to control room envelope habitability consistent with Technical Specification Task Force (TSTF) Traveler TSTF–448.

*Date of issuance:* April 16, 2008.

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

*Amendment No.:* 264.

*Facility Operating License No. DPR–50.* Amendment revised the license and the technical specifications.

*Date of initial notice in* ***Federal Register:*** June 5, 2007 (72 FR 31100). The supplements dated January 18, 2008, and March 14, 2008, 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 determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 2008.

*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:* January 22, 2007, as supplemented on June 21, July 18, July 31, and October 15, 2007, and January 24, February 14, March 5, and March 21, 2008.

*Brief description of amendments:* Change the Technical Specifications (TSs) to support the transition to AREVA fuel and core design methodologies.

*Date of issuance:* March 27, 2008.

*Effective date:* Date of issuance, to be implemented on Unit 1 prior to startup from the 2008 refueling outage, and to be implemented on Unit 2 prior to startup from the 2009 refueling outage.

*Amendment Nos.:* 246 and 274.

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

*Date of initial notice in* ***Federal Register:*** December 4, 2007 (72 FR 68208). The supplements dated January 24, February 14, March 5, and March 21, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* November 7, 2007.

*Brief description of amendment:* The amendment deletes License Condition 2.F, which requires reporting of violations of certain other requirements contained in Section 2.C of the license.

*Date of issuance:* April 15, 2008.

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

*Amendment No.:* 206.

*Facility Operating License No. NPF–21:* The amendment revised the Facility Operating License.

*Date of initial notice in* ***Federal Register:*** December 4, 2007 (72 FR 68211) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 2008.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* May 22, 2007.

*Brief description of amendment:* The amendment incorporates technical specification (TS) changes based on Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF)–497–A, ''Changes to Reflect Revision of 10 CFR 50.55a,'' Revision 0, as modified by NRC-approved TSTF–497, ''Limit Inservice Testing Program [Surveillance Requirements] SR 3.0.2 Application to Frequencies of Two years or Less.'' Specifically, the amendment revises Palisades Nuclear Plant TS Section 5.5.7, ''Inservice Testing Program,'' to update references to the American Society of Mechanical Engineers code and applicability of the provisions of SR 3.0.2.

*Date of issuance:* April 15, 2008.

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

*Amendment No.:* 232.

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

*Date of initial notice in* ***Federal Register:*** August 28, 2007 (72 FR 49575). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 2008.

*No significant hazards consideration comments received:* No.

*Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont*

*Date of application for amendment:* October 18, 2007.

*Brief description of amendment:* The amendment revised the Technical Specifications to change requirements related to emergency diesel generator (EDG) fuel oil tank volume, EDG fuel oil testing and reactor building crane inspections.

*Date of Issuance:* April 17, 2008.

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

*Amendment No.:* 231.

*Facility Operating License No. DPR–28:* Amendment revised the License and Technical Specifications.

*Date of initial notice in* ***Federal Register:*** December 18, 2007 (72 FR 71711).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 17, 2008.

*No significant hazards consideration comments received:* No.

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

*Date of amendment request:* April 24, 2007, as supplemented by electronic mail dated February 12, 2008.

*Brief description of amendment:* The change adds Optimized ZIRLO as an acceptable fuel rod cladding material in the Waterford Steam Electric Station, Unit 3, Technical Specification (TS) 5.3.1, ''Fuel Assemblies.'' TS 5.3.1 currently identifies, in part, Zircaloy or ZIRLO PM fuel rod cladding as the allowable fuel rod cladding material.

*Date of issuance:* April 16, 2008.

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

*Amendment No.:* 215.

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

*Date of initial notice in* ***Federal Register:*** May 22, 2007 (72 FR 28720). The supplemental electronic mail dated February 12, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 2008.

*No significant hazards consideration comments received:* No.

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

*Date of amendment request:* August 2, 2007, as supplemented by letters dated January 17, March 10, and electronic mail dated March 24, 2008. In addition, Entergy submitted for review and approval the revised emergency core cooling system (ECCS) performance analysis by letter dated August 9, 2007, as supplemented by letter dated January 21, 2008; and a supplement to the ECCS performance analysis by letter dated October 4, 2007, as supplemented by letter dated March 4, 2008.

*Brief description of amendment:* The changes to the technical specifications add new analytical methods and modify the containment average air temperature and safety injection tank level to support the implementation of Combustion Engineering 16 x 16 Next Generation Fuel (NGF) as defined in Westinghouse Topical Report WCAP–16500–P beginning in Cycle 16 commencing after the spring 2008 refueling outage.

*Date of issuance:* April 15, 2008.

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

*Amendment No.:* 214.

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

*Date of initial notice in* ***Federal Register:*** September 11, 2007 (72 FR 51858). The supplemental letters dated January 17, and March 10, 2008, and electronic mail dated March 24, 2008, for changes to the TSs; the supplemental letter dated January 21, 2008, for review and approval of the revised ECCS performance analysis; and the supplemental letter dated March 4, 2008, for review and approval of the supplement to the ECCS performance analysis, 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 April 15, 2008.

*No significant hazards consideration comments received:* No.

*Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station (Braidwood), Units 1 and 2, Will County, Illinois*

*Date of application for amendment:* February 25, 2008, as supplemented by letters dated March 27, 2008, and April 9, 2008.

*Brief description of amendment:* The amendments revise Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Program,'' and TS 5.6.9, ''Steam Generator (SG) Tube Inspection Report.'' For TS 5.5.9, the amendment replaces the existing alternate repair criteria in the provisions for SG tube repair criteria, during Braidwood, Unit 2, Refueling Outage 13 and the subsequent operating cycle. For TS 5.6.9, three new reporting requirements are added to the existing seven requirements for Braidwood Station (Braidwood), Unit 2. These changes only affect Braidwood, Unit 2; however, this action is docketed for Braidwood, Units 1 and 2, because the TS are common to both units.

*Date of issuance:* April 18, 2008.

*Effective date:* As of the date of issuance and shall be implemented prior to the return to service from Braidwood, Unit 2, spring 2008 Refueling Outage 13.

*Amendment Nos.:* Unit 1–150; Unit 2–150.

*Facility Operating License Nos. NPF–72 and NPF–77:* The amendment revised the TSs and License.

*Date of initial notice in* ***Federal Register:*** March 11, 2008 (73 FR 13029).

The March 27, 2008, and April 9, 2008, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 18, 2008.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendments:* February 29, 2008.

*Brief description of amendments:* The amendments revised the licensing basis of ice condenser ice fusion time, specifying conditions under which plant operation may proceed in less than 5 weeks after ice baskets have been reloaded.

*Date of issuance:* April 16, 2008.

*Effective date:* As of the date of issuance, and shall be implemented prior to Unit 1 entering Mode 4 at the end of the 2008 refueling outage.

*Amendment No.:* 303 (for DCCNP–1) and 286 (for DCCNP–2).

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

*Date of initial notice in* ***Federal Register:*** March 12, 2008 (73 FR 13253)

The Commission's related evaluation of the amendment is contained in a safety evaluation dated April 16, 2008.

*No significant hazards consideration comments received:* No.

*Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee*

*Date of application for amendment:* July 26, 2007, as supplemented by letters dated October 3 and December 21, 2007, and February 29, 2008.

*Brief description of amendment:* The proposed amendment would add a new reference to Technical Specification 6.9.1.14.a, which lists documents that have been approved by the U.S. Nuclear Regulatory Commission for use in determining the core operating limits. The new reference is the Areva NP, Inc., Topical Report EMF–2103P–A, ''Realistic Large Break LOCA [Loss-Of-Coolant Accident] Methodology for Pressurized Water Reactors.''

*Date of issuance:* April 10, 2008.

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

*Amendment No.* 311.

*Facility Operating License No. DPR–79:* Amendment revises the technical specifications.

*Date of initial notice in **Federal Register:*** August 28, 2007 (72 FR 49583). The supplemental letters dated October 3 and December 21, 2007, and February 29, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 2008.

*No significant hazards consideration comments received:* No.

**Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing**

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

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

*Arizona Public Service Company, et al., Docket No. STN 50–529, Palo Verde Nuclear Generating Station, Unit 2, Maricopa County, Arizona*

*Date of amendment request:* April 10, 2008.

*Brief Description of amendment request:* The proposed amendment would revise Technical Specification (TS) 3.5.5, Refueling Water Tank (RWT) to increase the minimum required RWT level indications and the corresponding borated water volumes in TS Figure 3.5.5–1, ''Minimum Required RWT Volume,'' by 3 percent. This change will ensure that there is adequate water volume available in the RWT to ensure that the engineered safety feature pumps and the new containment recirculation sump strainers will meet their design functions during loss-of-coolant accidents.

*Date of publication of individual notice in **Federal Register:*** April 17, 2008 (73 FR 20961).

*Expiration date of individual notice:* May 1, 2008.

Dated at Rockville, Maryland, this 28th day of April, 2008.

For the Nuclear Regulatory Commission.

**Catherine Haney,**

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

[FR Doc. E8–9679 Filed 5–5–08; 8:45 am]

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## NUCLEAR REGULATORY COMMISSION

**[Docket No. 50–133]**

## Environmental Assessment and Finding of No Significant Impact Related to Issuance of Exemption for the Humboldt Bay Power Plant Unit 3, License DPR–007, Humboldt, California

**AGENCY:** U.S. Nuclear Regulatory Commission.

**ACTION:** Environmental Assessment and Finding of No Significant Impact.

**FOR FURTHER INFORMATION CONTACT:** John Hickman, Division of Waste Management and Environmental Protection, Office of Federal and State Materials and Environmental Management Programs, U.S. Nuclear Regulatory Commission, Mail Stop: T8F5, Washington, DC 20555–0001. Telephone: (301) 415–3017; e-mail *john.hickman@nrc.gov.*

**SUPPLEMENTARY INFORMATION:**

### I. Introduction

The U.S. Nuclear Regulatory Commission (NRC) staff is considering a request dated November 5, 2007, by the Pacific Gas and Electric Company (PG&E or the Licensee), to approve a request for exemption from the requirements set forth in 10 CFR 50.54(p) and 10 CFR Part 73. The requested exemptions from the security requirements for Humboldt Bay Power Plant (HBPP) would be effective after the spent fuel has been removed from the reactor site by the licensee and relocated to the new Independent Spent Fuel Storage Installation (ISFSI).

This Environmental Assessment (EA) has been developed in accordance with the requirements of 10 CFR 51.21.

### II. Environmental Assessment

*Background*

HBPP was permanently shut down in July 1976, and until recently was in safe storage condition (SAFSTOR). SAFSTOR is defined as a method of decommissioning in which the nuclear facility is placed and maintained in safe condition for an extended period of time to permit radioactive material to decay to levels that facilitate subsequent decontamination and decommissioning of the facility. A decommissioning plan was approved in July 1988. Subsequent to the 1997 decommissioning rule, the licensee converted its decommissioning plan into its Defueled Safety Analysis Report which is updated every two years. A Post Shutdown Decommissioning Activities Report was issued by the licensee in February 1998. On September 2, 2005, the NRC approved the HB ISFSI Physical Security Plan (PSP) that PG&E submitted on July 11, 2005. On November 17, 2005, the NRC issued Materials License SNM–2514 for the HBPP ISFSI that included approval of the HBPP ISFSI PSP. In approving the Humboldt Bay ISFSI PSP, the NRC found that the plan meets the security requirements in 10 CFR Part 72 Subpart H, ''Physical Protection,'' meets the requirements in 10 CFR 73.51, ''Requirements for the Physical Protection of Stored Spent Nuclear Fuel and High-Level Radioactive Waste,'' and provides reasonable assurance that physical protection of the spent nuclear fuel stored at the ISFSI will not constitute an unreasonable risk to public health and safety. Currently, the licensee is maintaining the reactor security plan consistent with the requirements of 10 CFR Part 73 and 10 CFR 50.54(p). Contingent upon approval of the subject exemption and associated amendment, the ISFSI PSP will become effective upon the complete transfer of spent nuclear fuel from the spent fuel pool to the ISFSI.

*Proposed Action*

The proposed action would eliminate the security plan requirements for the 10 CFR Part 50 licensed site after the