1 p.m.—Meeting with the Advisory Committee on Reactor Safeguards (ACRS), (Contact: John Larkins, 301–415–7360).

This meeting will be webcast live at the Web address: *http://www.nrc.gov.*

Week of December 12, 2005—Tentative

Monday, December 12, 2005.

8:50 a.m.—Affirmation Session (Public Meeting) (Tentative), a. Exelon Generation Company, LLC (Early Site Permit for Clinton Site). (Tentative).

9 a.m.—Discussion of Security Issues (Closed—Ex. 1).

Wednesday, December 14, 2005.

1:30 p.m.—Discussion of Security Issues (Closed—Ex. 1).

Thursday, December 15, 2005.

1:30 p.m.—Briefing on Threat Environment Assessment (Closed—Ex. 1).

Week of December 19, 2005—Tentative

There are no meetings scheduled for the Week of December 19, 2005.

Week of December 26, 2005—Tentative

There are no meetings scheduled for the Week of December 26, 2005.

Week of January 2, 2006—Tentative

There are no meetings scheduled for the Week of January 2, 2006.

Week of January 9, 2006—Tentative

Tuesday, January 10, 2006.

9:30 a.m.—Briefing on International Research and Bilateral Agreements, (Contact: Roman Schaffer, 301–415–7606).

This meeting will be webcast live at the Web address: *http://www.nrc.gov.*

Wednesday, January 11, 2006.

9:30 a.m.—Meeting with Advisory Committee on Nuclear Waste (ACNW), (Contact: John Larkins, 301–415–7360).

This meeting will be webcast live at the Web address: *http://www.nrc.gov.*

Thursday, January 12, 2006.

9:30 a.m.—Discussion of Security Issues (Closed—Ex. 1 & 2).

*The schedule for commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. contact person for more information: Michelle Schroll, (301) 415–1662.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

Additional Information

The Affirmation Session tentatively scheduled on November 30, 2005, at 9:25 a.m. has been rescheduled tentatively on December 12, 2005, at 8:50 a.m.

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g., braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–415–2100, or by e-mail at *aks@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: December 1, 2005.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 05–23706 Filed 12–2–05; 11:00 am]

BILLING CODE 7590–01–M

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 to 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 November 9, 2005 to November 21, 2005. The last biweekly notice was published on November 22, 2005 (70 FR 70641).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives 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, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. 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 persons 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 a hearing or a petition for leave to intervene must be filed 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; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this 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.*

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

Date of amendments request: July 13, 2005.

Description of amendments request: The proposed amendment would revise Technical Specification (TS) 1.1, ''Definitions,'' TS 3.4.13, ''RCS [reactor coolant system] Operational Leakage,'' TS 5.5.9, ''Steam Generator Tube Surveillance Program,'' and TS 5.6.9, ''Steam Generator Tube Inspection Report,'' and add a new specification (TS 3.4.18) for Steam Generator (SG) Tube Integrity. The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting Technical Specification Task Force Change Traveller 449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated July 13, 2005.

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

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

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A SGTR [steam generator tube rupture] event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as MSLB [main steam line break], rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

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

Attorney for licensee: Carey Fleming, Sr. Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Branch Chief: Richard J. Laufer.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: October 31, 2005.

Description of amendment request: The proposed amendment change would add Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.8, to allow a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). In addition, a proposed change to LCO 3.0.1 is required to reference the addition of LCO 3.0.8.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated as TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated October 31, 2005.

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

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

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

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

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

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: David G. Pettinari, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Branch Chief: L. Raghavan.

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: October 3, 2005.

Description of amendment request: The proposed amendment revises Technical Specification (TS) Surveillance Requirements (SRs) to reflect changes to the Emergency Core Cooling System throttle valves. The proposed amendment will add the modified throttle valves to the surveillance, remove existing throttle valves that are now locked closed from the surveillance, and add existing valves to the surveillance that are used in a throttle position when open.

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 Surveillance Requirement (SR) 3.5.2.6 adds nine valves and removes two valves in the High Head Safety Injection (HHSI) system discharge lines. The SR requires verification that identified ECCS [emergency core cooling system] throttle valves position stops are in the correct position. The change reflects a stretch power uprate (SPU) modification that added throttle valves SI–2165, 2166, 2168, 2169, 2170, 2171, and 2172, and locked closed valves Sl-856A and 856F. This amendment is adding to the SR those throttle valves which are now under administrative control and deletes the valves which no longer perform a throttle function. The amendment also adds hot leg valves Sl-856B and 856G which are used as throttle valves but never included in the SR. Valve Sl-856G still performs a throttle function and valve SI–856B can still be considered a throttle valve when used to trim system resistance. Verification of valve position has no effect on the probability of an accident previously evaluated since the HHSI system is not associated with the initiation of any accident. The verification of valve positions that will be required by the revised SR provides additional assurance that the HHSI throttle valves are in the position that is established by flow testing. Providing assurance of required valve positions does not increase the consequences of an accident previously evaluated.Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change to Surveillance Requirement 3.5.2.6 adds nine valves and removes two valves in the High Head Safety Injection (HHSI) system discharge lines. The SR requires verification that identified ECCS throttle valves position stops are in the correct position. The change corrects a deficient surveillance and does not affect the function of the valves or otherwise affect the design and operation of plant systems and components and therefore no new accident scenarios would be created. Therefore, no new failure modes are being introduced that could lead to different accidents.

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

Response: No.

The proposed change to Surveillance Requirement 3.5.2.6 adds nine valves and removes two valves in the High Head Safety Injection (HHSI) system discharge lines. The SR requires verification that identified ECCS throttle valves position stops are in the correct position. The change reflects a stretch power uprate (SPU) modification that added throttle valves SI–2165, 2166, 2168, 2169, 2170, 2171, and 2172, and locked closed valves Sl-856A and 856F. The proposed amendment also adds valves SI–856B and 856G which are used as throttle valves but never included in the SR. Valve Sl-856G still performs a throttle function and valve Sl-856B can still be considered a throttle valve when used to trim system resistance. The frequency for verification of throttle valve stop positions is not altered by this amendment so this has no effect on the margin of safety. The valves for which verification of positions stops is required reflect the manner in which the system is currently analyzed and configured so the proposed change serves to maintain the required margin of safety by adding to the Technical Specifications the surveillances presently being administratively controlled. 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. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: June 29, 2005.

Description of amendment request: Entergy Operations, Incorporated (Entergy) proposes to relocate the License Condition associated with the Shutdown Cooling (SDC) Open Permissive Interlock (OPI) to the Technical Requirements Manual (TRM). The Nuclear Regulatory Commission (NRC) approved Standard Technical Specifications, Combustion Engineering Plants (NUREG–1432) include a surveillance requirement for this function due to the complexity and differences of plant designs, which would not support complete removal of the function from the NUREG. For Arkansas Nuclear One, Unit 2 (ANO–2), however, the OPI is not an assumed function that supports the accident analysis and does not meet the criteria in Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR) for inclusion in the technical specifications.

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 OPI function is not required to ensure continued safe operation of the ANO–2 facility. The OPI function is not considered an accident precursor or relied upon as a means of accident mitigation. The proposed change has no affect on plant design or operation.

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

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

Response: No.

The relocation of the OPI function to the TRM does not require any physical alteration to the plant or alter plant design. The OPI function is not considered an accident initiator nor is this function credited in any safety analyses for the prevention or mitigation of any accident.

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 OPI function is not credited in a margin of safety analysis for any accident previously evaluated. Relocation of the OPI function requirements will not result in a credible increase in nuclear safety risk. Appropriate alarm, design features, and administrative controls continue to ensure proper isolation of the SDC system during plant operations with elevated RCS [reactor cooling system] pressures. In addition, the OPI function will be relocated to the TRM, which is part of the Safety Analysis Report (SAR) and controlled by 10 CFR 50.59.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: September 19, 2005.

Description of amendment request: The proposed change will modify the Surveillance Requirements related to Arkansas One, Unit 2, technical specification (TS) 3.1.1.4, Moderator Temperature Coefficient (MTC), and will allow the use of topical report WCAP–16011-P-A, ''Startup Test Activity Reduction Program.'' A change to NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants,'' has been proposed in Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF–486 to incorporate the allowance to use WCAP–16011–P–A. The traveler was submitted for Nuclear Regulatory Commission (NRC) approval in June 2005.

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 MTC is not an initiator of any previously evaluated accidents. As an input into accident analyses, the MTC is used to predict plant behavior in the event of an accident. It was demonstrated in WCAP–16011-P-A that the modified MTC verification (*i.e.*, measured RCS [reactor coolant system] boron concentration) is adequate to ensure that the MTC remains within the limits provided the STAR applicability requirements are met. It was also demonstrated in WCAP–16011-P-A that the elimination of the EOC [emergency operations center] MTC measurement is acceptable when the applicability requirements given in WCAP–16011-P-A are met and the result of the MTC determination performed prior to reaching a Rated Thermal Power equilibrium boron concentration of 800 ppm is within a tolerance of $\pm 0.16 \times 10^{-4}$ Dk/k/°F from the corresponding design value.

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

2. Does the proposed change create the possibility of a new or different kind of

accident from any accident previously evaluated?

Response: No.

The proposed change does not involve a physical alteration to the plant (*i.e.*, no new or different type of structure, system, or component will be installed).

The methods governing normal plant operations are not altered by the 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 reduction in a margin of safety?

Response: No.

The proposed change will not affect the margin of safety. The MTC limits are unaffected and an acceptable method will be used to demonstrate that MTC is within its limits.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: September 19, 2005.

Description of amendment request: The proposed change will modify the ANO–2 technical specification (TS) 3.1.1.5, Minimum Temperature for Criticality. Specifically, the proposed change will raise the minimum temperature for criticality from the current value of ^{3} 525 °F to ^{3} 540 °F. Changes are also proposed to the Action statement and Surveillance Requirements to support the increase in temperature. The change is needed to support core design.

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.

There are no accident analyses that dictate the minimum temperature for criticality. The minimum temperature for criticality is not an accident initiator. It is used in the reload analyses as a limiting temperature at which the core design is verified to satisfy the limit of the positive moderator temperature coefficient (MTC) specified in the ANO–2 TS and Core Operating Limits Report (COLR). The minimum temperature for criticality is one of many input parameters used in the reload design analytical calculation that confirms the core design satisfies the MTC TS and COLR.

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 to increase the minimum temperature for criticality does not result in any plant design changes. In addition, the minimum temperature at which the reactor is taken critical is not an accident initiator. The nominal average reactor coolant system temperature during an approach to criticality is several degrees higher than the limit proposed for the minimum temperature for criticality.

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 increase of the minimum temperature for criticality in conjunction with the use of a sufficient number of burnable absorber rods, which will be incorporated into the core design, will ensure the current TS limits, as reflected in the COLR, for the most positive MTC will continue to be satisfied.

The current transient analysis results are bounding and remain applicable.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 25, 2005.

Description of amendment request: The proposed change will modify the Waterford 3 Technical Specification (TS) 3.1.1.4, Minimum Temperature for Criticality. Specifically, the proposed change will raise the minimum temperature for criticality from the current value of ≥520°F to ≥533°F. Changes are also proposed to the Action statement and Surveillance Requirements to support the increase in temperature.

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 minimum temperature for criticality is not an accident initiator. It is used in the reload analyses as a limiting temperature at which the core design is verified to satisfy the limit of the positive moderator temperature coefficient (MTC) specified in the Waterford 3 TS and Core Operating Limits Report (COLR). The minimum temperature for criticality is one of many input parameters used in the reload design analytical calculation that confirms the core design satisfies the MTC TS and COLR.

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 to increase the minimum temperature for criticality does not result in any plant design changes. In addition the minimum temperature at which the reactor is taken critical is not an accident initiator. The nominal average reactor coolant system temperature during an approach to criticality is several degrees higher than the limit proposed for the minimum temperature for criticality.

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 increase of the minimum temperature for criticality in conjunction with the appropriate core designs will ensure the current TS limits, as reflected in the COLR, for the most positive MTC will continue to be satisfied.

The current transient analysis results are bounding and remain applicable.

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: N. S. Reynolds, Esquire, Winston & Strawn, 1700 K Street NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 25, 2005.

Description of amendment request: The proposed change will modify the Surveillance Requirements (SRs) related to Waterford 3 Technical Specification (TS) 3.1.1.3, Moderator Temperature Coefficient (MTC) and will allow the use of the Startup Test Activity Reduction Program (WCAP–16011–P–A).

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

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

Response: No.

The MTC is not an initiator of any previously evaluated accidents. As an input into accident analyses, the MTC is used to predict plant behavior in the event of an accident. It was demonstrated in WCAP–16011–P–A that the modified MTC verification (i.e., measured RCS [reactor coolant system] boron concentration) is adequate to ensure that the MTC remains within the limits, provided the STAR applicability requirements are met. It was also demonstrated in WCAP–16011–P–A that the elimination of the EOC [end-of-cycle] MTC measurement is acceptable when the applicability requirements given in WCAP–16011–P–A are met and the result of the MTC determination performed at greater than 15 percent of Rated Thermal Power and prior to reaching 40 EFPD [effective full power days] is within a tolerance of $\pm 0.16 \times 10^{-4}$ Δk/k/°F from the corresponding design value.

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

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

Response: No.

The proposed change does not involve a physical alteration to the plant (i.e., no new or different type of structure, system, or component will be installed). The methods governing normal plant operations are not altered by the proposed TS change.

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

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

Response: No.

The proposed change will not affect the margin of safety. The MTC limits are unaffected and an acceptable method will be used to demonstrate that MTC is within its limits.

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: N. S. Reynolds, Esquire, Winston & Strawn, 1700 K Street NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: October 25, 2005.

Description of amendment request: The proposed change to Technical Specification 6.9.1.11, Core Operating Limits Report, will result in the addition of a methodology that will allow the use of zirconium diboride (ZrB_2) burnable absorber coating on fuel pellets.

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 will add topical report WCAP–16072–P–A to the NRC reviewed and approved analytical methods used to determine the core operating limits. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods.

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

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

Response: No.

The proposed change adds a reference to the topical report that allows the use of ZrB_2 as a burnable absorber coating on the fuel pellet. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods. This change is administrative in nature and does not create a new or different type of accident than previously evaluated because the design requirements for the facility remain the same.

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 add WCAP–16072–P–A to the list of referenced topical reports. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods.

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: N. S. Reynolds, Esquire, Winston & Strawn, 1700 K Street NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment requests: July 29, 2005.

Description of amendment requests: The proposed amendments would delete requirements from the Technical Specifications (TSs) to submit monthly operating reports and annual occupational radiation exposure reports. The changes are consistent with

Revision 1 of Nuclear Regulatory Commission (NRC) approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–369, ''Removal of Monthly Operating and Occupational Radiation Exposure Report.'' The availability of this TS improvement was announced in the **Federal Register** (69 FR 35067) on June 23, 2004, as part of the Consolidated Line Item Improvement Process (CLIIP).

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on April 29, 2004 (69 FR 23542). The licensee affirmed the applicability of the model NSHC determination in its application dated July 29, 2005.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC (which was previously published in 69 FR 23542) 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 eliminates the Technical Specifications reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the Technical Specification reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. 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.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based on the reasoning presented above and the previous discussion of the amendment request, the NRC staff proposes to determine that the requested change does not involve a significant hazards consideration.

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: L. Raghavan.

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

Date of amendment requests: August 10, 2005.

Description of amendment requests: The proposed amendments would delete the power range neutron flux high negative rate trip function from each unit's Technical Specifications. The licensee's proposed changes are based on the methodology presented in Westinghouse Topical Report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event,'' which had been previously accepted by the Nuclear Regulatory Commission staff.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

The removal of the power range neutron flux high negative rate trip function from technical specifications does not increase the probability or consequences of reactor core damage accidents resulting from dropped Rod Cluster Control Assembly (RCCA) events previously analyzed. The safety functions of other safety-related systems and components, which are related to mitigation of these events, [will] not [be] altered. All other Reactor Trip System and Engineered Safety Features Actuation Systems protection functions are not impacted by the elimination of the trip function. The dropped RCCA accident analysis does not rely on the negative flux rate trip to safely shut down the plant. The safety analysis of the plant is unaffected by the proposed change. Since the safety analysis is unaffected, the calculated radiological releases associated with the analysis are not 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 change does not adversely alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related systems or components. Nuclear Regulatory Commission (NRC)-approved Westinghouse Topical Report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event,'' dated January 1990 has demonstrated that the negative flux rate trip function can be eliminated.

Therefore, the proposed changes does not created 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 margin of safety associated with the acceptance criteria of any accident is unchanged. It has been demonstrated that the negative flux rate trip function can be eliminated by the NRC-approved methodology described in WCAP–11394–P–A. Donald C. Cook Nuclear Plant cycle-specific analyses have confirmed that for a dropped RCCA(s) event, limits on departure from nucleate boiling are not exceeded by eliminating the negative flux rate trip. The proposed change will have no [e]ffect on the availability, operability, or performance of safety-related systems and components.

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 requests involve no significant hazards consideration.

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: August 11, 2005.

Description of amendment request: The proposed change allows a delay time for entering a supported system Technical Specification (TS) when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of Paragraph 50.65(a)(4) of Title 10 of the *Code of Federal*

Regulations. Limiting Condition for Operation (LCO) 2.0.1(3) is added to the TS to provide this allowance and define the requirements and limitations for its use.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated August 11, 2005.

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

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

The proposed change allows a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 [LCO 2.0.1(3) for Fort Calhoun Station] are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8 [LCO 2.0.1(3)]. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG [Regulatory Guide] 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. [The proposed LCO 2.0.1(3) defines limitations on the use of the provision and includes a requirement for the licensee to assess and manage the risk associated with operation with an inoperable snubber.] The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Branch Chief: David Terao.

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

Date of amendment request: November 8, 2005.

Description of amendment request: The proposed amendment will modify Fort Calhoun Technical Specification (TS) 4.2.1, ''Fuel Assemblies,'' to permit the use of AREVA (Framatome ANP) M5™ advanced alloy for fuel rod cladding and structural components such as guide tubes, intermediate spacer grids, end plugs, and guide thimble tubes, beginning with Cycle 24. In addition, Omaha Public Power District proposes to modify TS 5.9 to include the Framatome ANP Topical Report evaluating the impact of M5™ material properties on NRC-approved methodology. M5™ is a proprietary, zirconium-based alloy that is a variant of Zr1Nb to replace zircaloy-4 in the construction of fuel assembly components.

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

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

Response: No.

The NRC[-]approved topical report BAW–10[2]27P–A (Reference 8.1 [of amendment request]) that provides the licensing basis for M5™ cladding and structural material, has shown that the M5™ alloy exhibits superior properties to the currently used zircaloy-4 material. The cladding by itself does not initiate an accident and therefore does not affect accident probability. It has been determined that M5™ cladding will not significantly affect the consequences of an accident.

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. 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 changes in the operation or overall configuration of the facility. Topical report BAW–10227P–A (Reference 8.1) demonstrated that the M5™ alloy will perform similar to or better than zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

Since the material properties of M5™ alloy are similar to or better than zircaloy-4, there will not be any significant change in the types of effluents that may be released off-site. There will not be any significant increase in occupational or public radiation exposure.

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.

AREVA has performed generic LOCA [loss-of-coolant accident] and non-LOCA evaluations and demonstrated the use of the M5™ material will have only a small, or beneficial, impact on the event consequences.

Plant-specific analyses using NRC-approved methodology for the mixed core will demonstrate that the reactor core safety limits will continue to be met.

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

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Branch Chief: David Terao.

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 requests: November 3, 2005.

Description of amendment requests: The proposed amendment revises Technical Specification (TS) Section 5.5.2.11 to modify the definitions of steam generator tube ''Repair Limit'' and ''Tube Inspection.'' The purpose of these changes is to define the extent of the required tube inspections and repair criteria within the tubesheet regions.

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.

This proposed change revises the San Onofre [Nuclear Generating Station,] Units 2 and 3 Technical Specifications (TS) by revising the definitions of steam generator ''Repair Limit'' and ''Tube Inspection[,]'' as contained in TS items 5.5.2.11.f.1.f and 5.5.2.11.f.1.h, respectively. This proposed change also adds words in the ''Operability determination'' requirement (item 5.5.2.11.f.2) to provide consistency with the proposed change in the definition of ''Repair Limit.'' These revisions maintain existing design limits and would not increase the probability or consequences of an accident involving tube burst or primary to secondary accident-induced leakage, as previously analyzed in the San Onofre [Nuclear Generating Station,] Units 2 and 3 Updated Final Safety Analysis Report (UFSAR). Also, the NEI 97–06 steam generator tube performance criterion for structural integrity and accident-induced leakage will continue to be satisfied.

Tube burst is precluded for a tube with defects within the tubesheet region because of the constraint provided by the tubesheet. As such, tube pullout resulting from the axial forces induced by primary to secondary differential pressures would be a prerequisite for tube burst to occur. An industry test program (WCAP–16208–P Revision 1), and follow-on San Onofre site-specific analysis (WCAP–16208–P Revision 1, Supplement 1) defined the non-degraded hot leg tube to tubesheet joint length and cold leg tube to tubesheet joint length required to preclude tube pullout and maintain acceptable primary to secondary accident-induced leakage, assuming that 100% [percent] of the steam generator tubes experienced complete circumferential separation (360 degree through wall crack) immediately below both the hot leg recommended inspection length (C*) and the cold leg C*. Any degradation below C* is shown by empirical test results and analyses to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event.

WCAP–16208–P Revision 1, with Supplement 1 includes a total 0.2 gpm [gallons per minute]/steam generator assumed value for primary to secondary accident-induced leakage. Inspection to the C* lengths will ensure that the postulated accident-induced leakage will remain below the current primary to secondary leakage assumption utilized in the UFSAR accident analyses (Chapter 15).

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.

Steam generator tube leakage and structural integrity will be maintained during all plant conditions upon implementation of the proposed inspection scope and repair limit changes to the San Onofre [Nuclear Generating Station,] Unit 2 and 3 Technical Specifications. These changes do not introduce any new mechanisms that might result in a different kind of accident from those previously evaluated. Even with the limiting circumstances of complete circumferential separation (360 degree through wall crack) of all of the tubes below the C* length, [a] tube pullout is precluded and leakage is predicted to be maintained within accident analysis assumptions.

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.

Operation with potential tube degradation below the C* inspection length within the tubesheet region of the steam generator tubing meets the intent of the inspection guidance of Regulatory Guide Number 1.83, Revision 1, titled Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes, the requirements of General Design Criteria 14, 15, 31 and 32 of 10 CFR 50, and the recommendations of the Nuclear Energy Institute in NEI 97–06, titled Steam Generator Program Guidelines.

The total leakage from an undetected flaw population below the C* inspection length under postulated accident conditions is accounted for to assure that it is within the bounds of the accident analysis assumptions. Adequate margin remains for other possible steam generator tube leak sources.

The proposed changes also maintain the structural and accident-induced leakage integrity of the steam generator tubes as required by NEI 97–06.

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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company,2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Branch Chief: David Terao.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: November 3, 2005.

Description of amendment request: The amendment would revise the Technical Specifications (TS) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment includes changes to the TS definition of Leakage, TS 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,'' TS 5.5.9, ''Steam Generator (SG) Program,'' TS 5.6.9, ''Steam Generator Tube Inspection Report,'' and adds TS 3.4.17, ''Steam Generator (SG) Tube Integrity.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97–06, ''Steam Generator Program Guidelines.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated November 3, 2005.

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

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

The proposed change requires an SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A steam generator tube rupture (SGTR) event is one of the design-basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of an SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design-basis accidents such as a main steamline break (MSLB), rod ejection, and reactor coolant pump locked rotor, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs are 1 gallon per minute or increases to 1 gallon per minute as a result of accident-induced stresses. The accident-induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design-basis accidents. The accident-induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design-basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design-basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design-basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design-basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of an SGTR accident, and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance-based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized-water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the 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.*

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

Date of application for amendments: July 9, 2004.

Brief description of amendments: The amendments revise the Operating Licenses and Technical Specifications (TSs) to allow operation of Palo Verde Nuclear Generating Station (PVNGS), Units 1 and 3 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94 percent above the current licensed power level of 3876 MWt. The proposed amendments would also make administrative changes to the PVNGS Unit 2 TSs so that the changed pages would apply to the three PVNGS units. Operation at the uprated power level with replacement steam generators has been approved for PVNGS Unit 2.

Date of issuance: November 16, 2005.

Effective date: November 16, 2005, and shall be implemented within 90 days of the date of issuance.

Amendment Nos.: Unit 1–157, Unit 2–157, Unit 3–157.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revise the Operating Licenses for Units 1 and 3 and the Technical Specifications for all three units.

Date of initial notice in **Federal Register:** September 28, 2004 (69 FR 57980). The June 2, June 3 (two letters), June 17, July 9 (two letters), July 19 (two letters), August 3, September 29, October 21, and November 1, 2005, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 31, 2005, as supplemented by letter dated September 13, 2005.

Brief description of amendment: The amendment permitted a one-time change to Technical Specification Table 3.3.8.1–1 to provide a one-time relaxation of the Loss of Power instrumentation requirements.

Date of issuance: September 15, 2005.

Effective date: As of the date of issuance to be implemented immediately.

Amendment No.: 147.

Facility Operating License No. NPF–47: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the NRC staff by the close of business on September 9, 2005. The notice was published in The St. Francisville Democrat (in St. Francisville) on September 8, 2005, and The Advocate (in Baton Rouge) on September 7, 2005. No public comments were received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Louisiana, and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 15, 2005.

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

Date of application for amendment: November 1, 2004, as supplemented by letters dated April 12, July 22, and September 26, 2005.

Brief description of amendment: The amendment authorizes the use of a single-failure-proof gantry crane for spent fuel cask handling operations up to 110 tons in weight.

Date of issuance: November 21, 2005.

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

Amendment No.: 244.

Facility Operating License No. DPR–26: The amendment allows use of the gantry crane for spent fuel cask handling operations up to 110 tons in weight.

Date of initial notice in ***Federal Register:*** December 7, 2004 (69 FR 70716). The April 12, July 22, and September 26, 2005, supplements 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 November 21, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 3, 2004, and its supplements dated February 24, June 23, and September 30, 2005.

Brief description of amendments: The amendments allow installation and use of a temporary cask pit spent fuel storage rack for Units 1 and 2. The cask pit rack would allow the storage of an additional 154 spent fuel assemblies for each unit. The total spent fuel pool storage capacity for each unit would be increased from the current 1324 spent fuel assemblies to 1478 assemblies for Cycles 14–16.

Date of issuance: November 21, 2005.

Effective date: As of the date of issuance, and shall be implemented upon installation of the temporary cask pit spent fuel rack.

Amendment Nos.: Unit 1—183; Unit 2–185.

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

Date of initial notice in ***Federal Register:*** December 21, 2004 (69 FR 76481). The February 24, June 23, and September 30, 2005, supplemental letters provided additional clarifying information, 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 November 21, 2005.

No significant hazards consideration comments received: Yes. The comments are addressed in the enclosure of the above Safety Evaluation.

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. 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 persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

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

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.

2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.

3. Miscellaneous—does not fall into one of the categories outlined above.

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

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

A request for a hearing or a petition for leave to intervene must be filed 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; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by email to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

PPL Susquehanna, LLC, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1 (SSES–1), Luzerne County, Pennsylvania

Date of amendment request: October 14, 2005, as supplemented on October 21 and November 2, 2005.

Description of amendment request: The amendment changed the SSES–1 Technical Specifications (TSs) by revising the SSES–1 Cycle 14 Minimum Critical Power Ratio Safety Limit in TS Section 2.1.1.2 from 1.08 to 1.09.

Date of issuance: November 10, 2005.

Effective date: November 10, 2005.

Amendment No.: 227.

Facility Operating License No. NPF–14: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. October 24, 2005 (70 FR 61475). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by December 22, 2005, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated November 10th 2005. The supplemental letters dated October 21 and November 2, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

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: Richard J. Laufer.

Virginia Electric and Power Company, Docket No. 50–338, North Anna Power Station, Unit No. 1 (North Anna 1), Louisa County, Virginia

Date of amendment request: November 3, 2005, as supplemented by letter dated November 4, 2005.

Description of amendment request: This amendment allows a temporary 7-day Completion Time to repair a weld leak that was discovered on the low-head safety injection (LHSI) suction pump piping. This change is needed to prevent an unnecessary plant transient and unscheduled shutdown of North Anna 1.

Date of issuance: November 4, 2005.

Effective date: As of the date of issuance and is applicable until the ''A'' train of the Unit 1 LHSI system is returned to operable status or until November 9, 2005, at 0330 hours, whichever occurs first.

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

Amendment No.: 246.

Renewed Facility Operating License No. NPF–4: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated November 4, 2005.

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: Evangelos Marinos.

Dated at Rockville, Maryland, this 28th day of November, 2005.

For the Nuclear Regulatory Commission.

Catherine Haney, Director,

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

[FR Doc. 05–23553 Filed 12–5–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Documents Regarding Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

FOR FURTHER INFORMATION CONTACT: Allen Hansen, Thermal Engineer, Criticality, Shielding and Heat Transfer Section, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20005–0001. Telephone: (301) 415–1390; fax number: (301) 415–8555; e-mail: *agh@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

Under contract with the Nuclear Regulatory Commission (NRC), the Pacific Northwest National Laboratory prepared the draft NUREG/CR–6886 report, ''Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire (BTF) Scenario.'' The BTF was chosen for the study because it represents a severe historical accident, even though it is a very low frequency event. This NUREG/CR documents the thermal analyses of three different spent fuel transportation packages exposed to the BTF scenario: Transnuclear's TN–68, Holtec's HI-STAR 100 and the NAC's LWT.

To date comments have been received from the State of Nevada, Office of the Governor, Agency For Nuclear Projects and the Western Interstate Energy Board. These comments do not need to be re-submitted.

The format of this NUREG/CR has been modified since original posting on the NRC Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html* in September 2005. The modified draft NUREG/CR is now posted on the NRC Web site at the following URLs:

http://www.nrc.gov/reading-rm/doc-collections/nuregs/docs4comment.html.

http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6886/.

These links include access to the formal comment template.

The results of this study strongly indicate that neither spent nuclear fuel (SNF) particles nor fission products would be released from a spent fuel shipping cask involved in a severe tunnel fire such as the Baltimore Tunnel Fire. None of the three cask designs analyzed for the Baltimore Tunnel fire scenario experienced internal temperatures that would result in rupture of the fuel cladding. Therefore, the radioactive material (*i.e.*, SNF particles or fission products) would be retained within the fuel rods.

For two of the casks, the TN–68 and the NAC–LWT, the maximum temperatures experienced in the regions of the lid, vent and drain ports exceeded the seals' rated service temperatures, making it possible to get a small release from the CRUD [1] that might spall off of the surfaces of the fuel rods. However, any release is expected to be very small due to a number of factors. These include: (1) The tight clearances maintained between the lid and cask body; (2) the low pressure differential between the cask interior and the outside; (3) the tendency of the small clearances to plug; and (4) the tendency of CRUD particles to settle or plate out. The potential releases calculated in Chapter 8 for the TN–68 rail cask and the NAC–LWT truck cask indicate that the release of CRUD from either cask, if any, would be very small. There would be no release from the HI–STAR 100 because the inner welded canister remains leak tight.

II. Summary

The purpose of this notice is to provide the public an opportunity to review and comment on the Draft NUREG/CR–6886 thermal analyses, the consequence analyses and the conclusions.

III. Further Information

The draft NUREG/CR can also be viewed at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* From this site you can access the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS accession number for the edited (format only) NUREG is ML053200024. This file is in ''black and white.'' The original draft is in color and can be accessed at the following accession numbers:

NUREG/CR Files	ADAMS accession No.
Spent Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario	ML052500391
Appendix A—Material Properties for COBRA–SFS Model of TN–68 Package	ML052490246
Appendix B—Material Properties for ANSYS Model of HI–STAR 100 Package	ML052490258
Appendix C—Material Properties for ANSYS Model of Legal Weight Truck Package	ML052490264
Appendix D—Blackbody View Factors for COBRA–SFS Model of TN–68 Package	ML052490268
Appendix E—HOLTEC HI–STAR 100 Component Temperature Distributions	ML052490270

[1] CRUD is an abbreviation of Chalk River Unknown Deposit, a generic term for various residues deposited on fuel rod surfaces, originally coined by Atomic Energy of Canada, Ltd. to describe deposits observed on fuel removed from the test reactor at Chalk River.