The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Antonio F. Dias (Telephone: 301/415–6805) between 8:15 a.m. and 5 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 8:15 a.m. and 5 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: November 28, 2006.

Michael R. Snodderly,

Branch Chief, ACRS/ACNW.

[FR Doc. E6–20515 Filed 12–4–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of December 4, 11, 18, 25, 2006, January 1, 8, 2007.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of December 4, 2006

Wednesday, December 6, 2006

2:45 p.m. Discussion of Security Issues (Closed—Ex. 1).

Thursday, December 7, 2006

9:25 a.m. Affirmation Session (Public Meeting) (Tentative) a. Hydro Resources, Inc. (Crownpoint, NM) Intervenors' Petition for Review of LBP–06–19 (Final Partial Initial Decision—NEPA Issues) (Tentative).

9:30 a.m. Discussion of Management Issues (Closed—Ex. 2).

Week of December 11, 2006—Tentative

Monday, December 11, 2006

1:30 p.m. Briefing on Status of Decommissioning Activities (Public Meeting) (Contact: Keith McConnell, 301–415–7295).

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

Tuesday, December 12, 2006

9:30 a.m. Briefing on Threat Environment Assessment (Closed—Ex. 1).

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1 & 3).

Wednesday, December 13, 2006

9:30 a.m. Briefing on Status of Equal Employment Opportunity (EEO) Programs (Public Meeting) (Contact: Barbara Williams, 301–415–7388).

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

Thursday, December 14, 2006

9:25 a.m. Affirmation Session (Public Meeting) (Tentative) a. Entergy Nuclear Vermont Yankee, LLC, & Entergy Nuclear Operations, Inc. (Vermont Yankee Nuclear Power Station), LBP–06–20 (Sept. 22, 2006), reconsid'n denied (Oct. 30, 2006) (Tentative).

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

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

Week of December 18, 2006—Tentative

There are no meetings scheduled for the Week of December 18, 2006.

Week of December 25, 2006—Tentative

There are no meetings scheduled for the Week of December 25, 2006.

Week of January 1, 2007—Tentative

There are no meetings scheduled for the Week of January 1, 2007.

Week of January 8, 2007—Tentative

Wednesday, January 10, 2007

9:30 a.m. Briefing on Browns Ferry Unit 1 Restart (Public Meeting) (Contact: Catherine Haney, 301–415–1453).

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

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* 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.

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

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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, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@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: November 30, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–9535 Filed 11–31–06; 10:04 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission 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, 2006, to November 21, 2006. The last biweekly notice was published on November 21, 2006 (71 FR 67391).

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, Rulemaking, Directives and Editing Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, 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: Rulemaking 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.*

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

Date of amendment request: April 11, 2006

Description of amendment request: The proposed amendment 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. The proposed changes are consistent with approval of TS Task Force (TSTF) change TSTF–372, Revision 4, ''Addition of LCO 3.0.8, Inoperability of Snubbers.''

The NRC staff issued a notice of availability of a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on November 24, 2004 (69 FR 68412).

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 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 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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: June 5, 2006.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) to clarify Surveillance Requirement (SR) 3.8.1.13 and its associated Bases to state that the SR only verifies that non-emergency diesel generator (DG) trips are bypassed. It is based upon, and consistent with, Industry Technical Specification Task Force (TSTF), Standard Technical Specification Traveler, TSTF–400–A, Revision 1, ''Clarify Surveillance Requirement on Bypass of DG Automatic Trips.''

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. Would implementation of the changes proposed in this LAR (License Amendment Request) involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR clarifies the purpose of Surveillance Requirement (SR) 3.8.1.13, which is to verify that non-emergency automatic diesel generator (DG) trips are bypassed in an accident. The DG automatic trips and their bypasses are not initiators of any accident that has been previously evaluated. Therefore, the probability of any of these accidents is not significantly increased. The function of the DG in mitigating accidents is not changed. The revised SR continues to ensure that the DG will operate as assumed in the accident analyses. Therefore, the consequences of any accident previously evaluated are not affected as well.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The changes proposed in this LAR only clarify the purpose of SR 3.8.1.13, which is to verify that non-emergency automatic DG trips are bypassed in an accident. The proposed change does not involve a physical change to the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation or testing. Thus, the changes proposed in this LAR do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The changes proposed in this LAR only clarify the purpose of SR 3.8.1.13, which is to verify that non-emergency automatic DG trips are bypassed in an accident. These changes clarify the purpose of the SR, which is to verify that the DG is capable of performing its assumed safety function. The safety function of the DG is unaffected, so the changes do not affect the margin of safety.

Therefore, this LAR does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: April 11, 2006.

Description of amendment request: The proposed amendment 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. The proposed changes are consistent with approval of TS Task Force (TSTF) Change TSTF–372, Revision 4, ''Addition of LCO 3.0.8, Inoperability of Snubbers.''

The NRC staff issued a Notice of Opportunity to Comment of a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on November 24, 2004 (69 FR 68412).

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 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 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: Ms. Lisa F. Vaughn, Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: April 11, 2006.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) related to steam generator (SG) tube integrity. The changes are consistent with the consolidated line-item improvement process (CLIIP), Nuclear Regulatory Commission-approved Revision 4 to Technical Specification Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.''

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 (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 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, 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 1–131 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 1–131 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 0.27 gallons per minute with no more than 135 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT 1–131 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 (main steamline break), 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.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: June 5, 2006.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) to clarify Surveillance Requirement (SR) 3.8.1.13 and its associated Bases to state that the SR only verifies that non-emergency diesel generator (DG) trips are bypassed. It is based upon, and consistent with, Industry Technical Specification Task Force (TSTF), Standard Technical Specification Traveler, TSTF–400–A, Revision 1, ''Clarify Surveillance Requirement on Bypass of DG Automatic Trips.''

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. Would implementation of the changes proposed in this LAR (License Amendment Request) involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR clarifies the purpose of Surveillance Requirement (SR) 3.8.1.13, which is to verify that non-emergency automatic diesel generator (DG) trips are bypassed in an accident. The DG automatic trips and their bypasses are not initiators of any accident that has been previously evaluated. Therefore, the probability of any of these accidents is not significantly increased. The function of the DG in mitigating accidents is not changed. The revised SR continues to ensure that the DG will operate as assumed in the accident analyses. Therefore, the consequences of any accident previously evaluated are not affected as well.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The changes proposed in this LAR only clarify the purpose of SR 3.8.1.13, which is to verify that non-emergency automatic DG trips are bypassed in an accident. The proposed change does not involve a physical change to the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation or testing. Thus, the changes proposed in this LAR do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The changes proposed in this LAR only clarify the purpose of SR 3.8.1.13, which is to verify that non-emergency automatic DG trips are bypassed in an accident. These changes clarify the purpose of the SR, which is to verify that the DG is capable of performing its assumed safety function. The safety function of the DG is unaffected, so the changes do not affect the margin of safety. Therefore, this LAR does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: July 31, 2006.

Description of amendment request: The proposed amendments would revise Technical Specification Section 3.6.3, ''Containment Isolation Valves,'' and its associated Bases, by removing the allowance to open the upper containment purge isolation valves in the applicable modes consistent with the lower containment purge isolation valves.

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

1. Does this LAR [License Amendment Request] involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The Containment Purge System is not capable of initiating any accident by itself so there will be no increase in the probability of an accident. Since these containment isolation valves will be maintained in the sealed closed position, there can be no increase in the consequences of an accident. The design and operation of the Containment Purge System is not being modified by this LAR. Therefore, approval and implementation of this LAR will have no effect on accident probabilities or consequences.

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

No. This LAR does not involve any physical changes to the Containment Purge System so no new or different accident causal mechanisms will be generated. Also, no changes are being made to the way in which the Containment Purge System is operated. Some surveillance tests will no longer be performed but these tests are no longer necessary since the affected components remain in their safe, design basis position. Consequently, plant accident analyses will not be affected by this LAR.

3. Does this LAR involve a significant reduction in a margin of safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be affected by the proposed changes. The containment isolation valves in the Containment Purge System will continue to perform their design basis function after this LAR is implemented.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: November 1, 2006.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements for inoperable snubbers by adding Limiting Condition of Operation (LCO) 3.0.8.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments to revise the plant-specific TS 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 that is in place for complying with the requirements of 10 CFR 50.65(a)(4). LCO 3.0.8 was proposed to be added to an individual TS providing this allowance, 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 model NSHC determination in its application dated November 1, 2006.

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 [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 net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a no significant hazards consideration.

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

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

NRC Branch Chief: David Terao.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: November 1, 2006.

Description of amendment request: The proposed change will revise the Grand Gulf Nuclear Station (GGNS), Unit 1, Technical Specification (TS) Surveillance Requirement 3.3.1.1.7 for the surveillance interval of the local power range monitor (LPRM) calibrations from 1,000 megawatt-days/ton (MWD/T) (approximately every 36 days) to 2,000 MWD/T (approximately every 72 days).

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 extended surveillance interval continues to ensure that the LPRM detectors are adequately calibrated to provide an accurate indication of core power distribution and local power changes. The change will not alter the basic operation of any process variables, structures, systems, or components as described in the safety analyses, and no new equipment is introduced. Hence, the probability of accidents previously evaluated is unchanged.

The thermal limits established by safety analysis calculations ensure that reactor core operation is maintained within fuel design limits during any Anticipated Operational Occurrence (AOO). The analytical methods and assumptions used in evaluating these transients and establishing the thermal limits assure adequate margins to fuel design limits are maintained. These methods account for various calculation uncertainties including radial bundle power uncertainty which can be affected by LPRM accuracy. Extending the LPRM calibration interval does not impact the existing uncertainties assumed in the GGNS safety analyses. Plant specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration interval does not affect the radial bundle power distribution uncertainty value currently used in the safety analysis. Hence the safety analysis calculations and the associated thermal limits are not affected by the extended LPRM calibration interval and the consequences of an accident previously evaluated are not changed.

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 TS amendment will not change the design function, reliability, performance, or operation of any plant systems, components, or structures. It does not create the possibility of a new failure mechanism, malfunction, or accident initiators not considered in the design and licensing bases. Plant operation will continue to be within the core operating limits that are established using NRC approved methods that are applicable to the GGNS design and the GGNS fuel.

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 thermal limits established by safety analysis calculations ensure that reactor core operation is maintained within fuel design limits during any Anticipated Operational Occurrence (AOO). The analytical methods and assumptions used in evaluating these transients and establishing the thermal limits assure adequate margins to fuel design limits are maintained. These methods account for various calculation uncertainties including radial bundle power uncertainty which can be affected by LPRM accuracy. Extending the LPRM calibration interval does not impact the existing uncertainties assumed in the GGNS safety analyses. Plant specific evaluation of LPRM sensitivity to exposure has determined that the extended calibration interval does not affect the radial bundle power distribution uncertainty value currently used in the safety analyses. The thermal limits determined by NRC approved analytical methods will continue to provide adequate margin to fuel design 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: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213

NRC Branch Chief: David Terao

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

Date of amendment request: March 1, 2006

Description of amendment request: The proposed amendment would modify the Special Operations Limiting Condition for Operation (LCO) 3.10.1, ''System Leakage and Hydrostatic Testing Operation,'' allowance for operation with the average reactor coolant temperature greater than 212 °F while considering operational conditions to be in MODE 4, to include operations where temperature exceeds 212 °F as a consequence of maintaining reactor pressure for a system leakage or hydrostatic test, or as a consequence of maintaining reactor pressure for control rod scram time testing initiated in conjunction with a system leakage or hydrostatic test. This change would allow more efficient testing during a refueling outage.

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

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

Response: No.

Technical Specifications currently allow for operation at >212 °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact the probability or 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 amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Technical Specifications currently allow for operation at >212 °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. No new operational conditions beyond those currently allowed by LCO 3.10.1 are introduced. The extended allowances would result from operations that commence at reduced temperatures, but approach the normal MODE 4 limit of 212 °F prior to completion of the inspections or testing. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

Response: No.

Technical Specifications currently allow for operation at >212 °F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact any margin of safety. Allowing completion of inspections and testing and supporting completion of scram time testing initiated in conjunction with a system leakage or hydrostatic test prior to power operation, results in enhanced safe operations by eliminating unnecessary maneuvers to control reactor temperature and pressure.

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

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

Attorney for licensee: Mr. R. E. Helfrich, Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: L. Raghavan.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: October 10, 2006.

Description of amendment requests: The amendment application proposes a revision to the Technical Specification Surveillance Requirement 4.1.1.3 to extend the containment airlock surveillance frequency from once per year to once every five years.

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

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated? No.

The proposed change does not introduce any new degradation or failure mechanism. The failure mechanism in this case would be a failure of an airlock door to open, thus no new release path to the environment is created. As no release path is created, there is not the possibility of a significant increase in the probability or consequences of an accident.

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? No.

The proposed change does not introduce any new degradation or failure mechanism.

The failure mechanism in this case would be a failure of an airlock door to open, thus no new release path to the environment is created. As no release path is created, there is not the possibility of a new or different kind of accident from any accident previously evaluated being created.

(3) Does the proposed change involve a significant reduction in a margin of safety? No.

The proposed change does not introduce any new degradation or failure mechanism. The failure mechanism in this case would be a failure of an airlock door to open, thus no new release path to the environment is created. Thus, 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.

NRC Branch Chief: Claudia Craig.

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

Date of amendment request: October 17, 2006.

Description of amendment request: The proposed amendment would revise the Cooper Nuclear Station (CNS) Technical Specifications (TS) 4.3.1.1.c by adding a new nominal center-to-center distance between fuel assemblies for the new storage racks, and would revise TS 4.3.3 by increasing the capacity of the spent fuel storage pool from 2366 assemblies to 2651 assemblies.

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

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

Response: No.

The probability of a seismic event, and the resulting loss of spent fuel pool cooling flow, is not influenced by the proposed changes. In addition, the probability of an accidental fuel assembly drop or misloading is primarily influenced by the methods used to lift and move these loads. The method of handling fuel will not be changed since the same equipment and procedures will be used. Shipping cask movements in the SFP [spent fuel pool] will not be performed during installation of the new racks. There is no change to the methods or equipment to be used in moving fuel casks. Expanding the spent fuel storage capacity does not have a significant impact on the frequency of occurrence for any accident previously evaluated.

Therefore, this change will not significantly increase the probability of occurrence of any accident previously analyzed.

The consequences of a dropped spent fuel assembly in the SFP have been re-evaluated for the proposed change by analyzing a potential impact onto the new racks. The results show that the postulated accident of a fuel assembly striking the new storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin required by the current TS (i.e., neutron multiplication factor [keff] less than or equal to 0.95) will be maintained. The structural damage to the Reactor Building, pool liner, and fuel assembly resulting from a dropped fuel assembly striking the pool floor or another assembly located in the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed modification, the postulated structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will not be increased since no changes are being made to in-core hold time or burnup as a result of the proposed amendment.

Loss of SFP cooling was evaluated. The concern with this event is a reduction of spent fuel pool water inventory as a result of boiling in the fuel pool, with the inventory reduction resulting in an unacceptable increase in dose rates. Loss of spent fuel pool cooling at CNS is mitigated procedurally by supplying makeup water to the pool prior to the time that the temperature of the pool reaches boiling. The thermal-hydraulic analysis for the proposed license amendment determined, for a complete loss of forced cooling and a full core discharge, that the minimum time to boil is 4.19 hours. This has been determined to be sufficient time for the operators to provide alternate means of makeup water to the SFP before the water begins to boil. Based on this the consequences of a loss of SFP cooling are not significantly increased.

The consequences of a design basis seismic event are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The new racks have been analyzed in their new configuration and were found to be safe during seismic motion. Fuel has been determined to remain intact and the storage racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the anticipated combinations of dead weight, thermal, and seismic loads. The Reactor Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Therefore, the consequences of a design basis seismic event are not increased.

The consequence of a fuel misloading accident has been analyzed for the worst possible storage configuration subsequent to the proposed modification. It has been determined that the consequences remain acceptable with respect to the same criteria used previously.

Therefore, the proposed change does not result in a significant increase in the consequences of a previously evaluated accident.

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

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

Response: No.

A drop of a fuel assembly onto fuel assemblies stored in the SFP has been previously analyzed for CNS and is not a new or different kind of accident. The only event which would represent a new or different kind of accident is an accidental drop of a rack during movement in the pool.

Dropping a rack onto stored spent fuel or the pool floor liner, commonly referred to as a ''heavy load drop,'' is not postulated due to the defense-in-depth approach to be taken. A lifting rig designed to meet the requirements of NUREG 0612 [Nuclear Regulatory Commission technical report designation 0612] and ANSI N 14.6 [American National Standards Institute N 14.6] will be used to install the new racks. Dropping a new rack onto fuel is precluded by not allowing the new racks being placed into the SFP to travel over racks containing fuel assemblies. A rack drop to the pool liner is not postulated since the lifting components either provide redundancy in supporting the racks or are designed with safety margins greater than a factor of ten. Movements of heavy loads over the pool will comply with the applicable administrative controls and guidelines (i.e. plant procedures, NUREG 0612, etc.). Therefore, the rack drop does not represent a new or different kind of accident.

The proposed change does not alter the operation of the plant or equipment credited for the mitigation of the design basis accidents. The proposed change does not affect the important parameters required to ensure safe fuel storage.

In summary, 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 amendment involve a significant reduction in a margin of safety?

Response: No.

The function of the spent fuel pool is to store the fuel assemblies in a subcritical and coolable configuration under postulated environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design meets the applicable requirements for safe storage and is functionally compatible with the SFP.

The Holtec Licensing Report was prepared using the guidance of the applicable provisions of the NRC Guidance entitled, ''OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications.'' The rack materials used are compatible with the spent fuel assemblies and the SFP environment. The design of the new racks preserves the proper margin of safety during abnormal loads, e.g., loads from a seismic event, a dropped assembly, and tensile loads from a stuck fuel assembly. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

The methodology used in the criticality analysis of the expanded spent fuel pool complies with the appropriate NRC guidelines and the ANSI standards (Draft GDC 66 [General Design Criterion 66], NUREG 0800, Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, and ANSI ANS 8.17 [American Nuclear Society 8.17]).

The subcriticality margin (k_{eff}) for spent fuel stored in the SFP is required to be less than or equal to 0.95 under normal storage, fuel handling, and accident conditions, including uncertainties. This margin will be maintained with the proposed increased capacity.

The thermal-hydraulic and cooling evaluation of the pool determined that the pool can be maintained below the specified thermal limits under the conditions of the maximum heat load. The pool temperature will not exceed the design temperature of 150°F during operation of the cooling systems. The maximum local water temperature in the hot channel will remain below the boiling point. The maximum cladding temperature after a loss of cooling remains less than the current licensing basis value of 350 °F with bulk boiling in the pool. The stored fuel will not undergo any significant heat up with blockage of a dropped fuel assembly lying horizontally on top of the racks. The thermal limits specified for the evaluations performed to support the proposed change are the same as those which were used in the previous evaluations.

The time to boiling, in the event of a complete loss of SFP cooling with a full core discharge, has been reduced from 5 hours to 4.19 hours. However, this has been determined to be sufficient time for providing makeup to the SFP.

Based on the above it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: David Terao.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: October 19, 2006.

Description of amendment request: The proposed amendment would revise the surveillance requirements in Technical Specification (TS) 4.1.1, ''Control Rod System,'' to modify the conditions under which scram time testing (STT) of control rods is required, and add a requirement to perform STT on a defined portion of control rods, at a specified frequency, during the operating cycle. The requirement to test ''eight selected [control] rods'' after a reactor scram or other outage would be replaced by a requirement to periodically test at least 20 control rods, on a rotating basis, every 180 days.

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

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

Response: No.

The proposed change adds new surveillance requirements (SR) to the MCPR [minimum critical power ratio] Technical Specification (TS) which requires determination of the MCPR operating limit following the completion of scram time testing (STT) of the control rods. Use of the scram speed in determining the MCPR operating limit (i.e., Option B) is an alternative to the current method for determining the operating limit (i.e., Option A). The probability of an accident previously evaluated is unrelated to the MCPR operating limit that is provided to ensure no fuel damage results during anticipated operational occurrences. This is an operational limit to ensure conditions following an assumed accident do not result in fuel failure and therefore do not contribute to the occurrence of an accident.

The proposed change revises allowable conditions for the STT of non-maintenance affected control rods and eliminates the requirement to test ''eight [selected] rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. No active or passive failure mechanisms that could lead to an accident are affected by this proposed change and the STT acceptance criteria are not being revised. Therefore, the proposed change in STT requirements does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change ensures that the appropriate MCPR operating limit is in place. By implementing the correct MCPR operating limit, the MCPR SL [safety limit] will continue to be ensured. Ensuring the MCPR SL is not exceeded will result in prevention of fuel failure. Therefore, since there is no increase in the potential for fuel failure, there is no increase in the consequences of any accidents 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 new SR to the MCPR TS which requires determination of the MCPR operating limit following the completion of the [STT] of the control rods. The proposed change revises allowable conditions for the STT of non-maintenance affected control rods and eliminates the requirement to test ''eight [selected] rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. The proposed change does not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. No new failures have been created by the addition of the proposed SR and the use of the alternate method for determining the MCPR operating limit. 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.

Use of Option B for determining the MCPR operating limit will result in a reduced operating limit in comparison to the use of Option A. However, a reduction in the operating limit margin does not result in a reduction in the safety margin. The MCPR SL remains the same regardless of the method used for determining the operating limit. The proposed change revises allowable conditions for the STT of non-maintenance affected control rods and eliminates the requirement to test ''eight [selected] rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. No active or passive failure mechanisms that could adversely impact the consequences of an accident are affected by this proposed change. All analyzed transient results remain within the design values for structures, systems and components. 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: October 23, 2006.

Description of amendment request: The proposed changes to the technical specifications (TSs) would eliminate the use of the defined term CORE ALTERATIONS in the TSs.

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. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change eliminates the use of the defined term CORE ALTERATIONS from the Technical Specifications. CORE ALTERATIONS are not an initiator of any accident previously evaluated except a fuel handling accident. The revised Technical Specifications that protect the initial conditions of a fuel handling accident also require the suspension of movement of irradiated fuel assemblies, which protects the initial condition of a fuel handling accident.

Therefore, suspension of CORE ALTERATIONS do not affect the initiators of the accidents previously evaluated and suspension of CORE ALTERATIONS does not affect the mitigation of the accidents previously evaluated.

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical modification of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions.

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Only two accidents are postulated to occur during plant conditions where CORE ALTERATIONS may be made: A fuel handling accident and a boron dilution accident. Suspending movement of irradiated fuel assemblies prevents a fuel handling accident. Also, requiring the suspension of CORE ALTERATIONS is redundant to suspending movement of irradiated fuel assemblies and does not increase the margin of safety. CORE ALTERATIONS have no effect on a boron dilution accident. Core components are not involved in the initiation or mitigation of a boron dilution accident. Therefore, CORE ALTERATIONS have no effect on the margin of safety related to a boron dilution accident.

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: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Acting Branch Chief: L. Raghavan.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: August 4, 2006.

Description of amendment request: The amendments would allow the use of blind flanges for containment isolation in the containment purge system supply and exhaust lines, and make corresponding changes to the Technical Specifications (TSs). The amendments would also consolidate the containment isolation requirements by moving the requirements of TS 3/4 6.1.7, ''Containment Ventilation System,'' to TS 3/4 6.3.1 (TS 3/4 6.3 for Unit No. 2), ''Containment Isolation Valves.''

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

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

Response: No.

The proposed change to the Containment purge supply and exhaust penetrations presents no change in the probability or the consequence of an accident, since the penetrations continue to conform to the TS requirements for containment integrity, and will be appropriately tested as required by 10 CFR 50 Appendix J. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program. Containment integrity is not lessened by this change.

The change to the Containment Purge System does not affect the design basis limit for any fission product barrier.

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 the Containment purge supply and exhaust penetrations does not change the function of the system and does not alter containment integrity. The penetrations continue to conform to the TS requirements for containment integrity and will be appropriately tested as required by 10 CFR 50 Appendix J. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

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

Response: No.

The proposed change will not alter any assumptions, initial conditions or results specified in any accident analysis. The Containment purge supply and exhaust penetrations will continue to conform to the TS requirements for containment integrity, and will be appropriately tested as required by 10 CFR 50 Appendix J. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program. Containment integrity is not lessened by this change. Therefore, there is no 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Harold K. Chernoff.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: October 3, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) and licensing basis to support the resolution of the Nuclear Regulatory Commission's (NRC's) Generic Safety Issue (GSI) 191, assessment of debris accumulation on containment sump performance and its impact on emergency recirculation during an accident, and NRC Generic Letter (GL) 2004–02.

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

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

Response: No.

The proposed changes include a physical alteration to the RS system to start the inside and outside [Recirculation Spray] RS pumps on [Refueling Water Storage Tank] RWST Level Low coincident with High High containment pressure. The RS system is used for accident mitigation only, and changes in the operation of the RS system cannot have an impact on the probability of an accident. The other changes do not affect equipment and are not accident initiators. The RWST Level Low instrumentation will comply with all applicable regulatory requirements and design criteria (e.g., train separation, redundancy, and single failure). Therefore, the design functions performed by the RS system are not changed.

Delaying the start of the RS pumps creates more challenging long-term containment pressure and temperature profiles. The environmental qualification of safety-related equipment inside containment was confirmed to be acceptable, and accident mitigation systems will continue to operate within design temperatures and pressures. Delaying the RS pump start reduces the emergency diesel generator loading early during a design basis accident, and staggering the RS pump start avoids overloading on each emergency bus. The reduction in iodine removal efficiency during the delay period is offset by changes to other assumptions in the [loss-of-coolant accident] LOCA dose analysis. The predicted offsite doses and control room doses following a design basis LOCA remain within regulatory limits.

The [Updated Final Safety Analysis Report] UFSAR safety analysis acceptance criteria continue to be met for the proposed changes to the RS pump start method, the proposed TS containment air partial pressure limits, the proposed TS containment temperature limit, the implementation of the GOTHIC containment analysis methodology, the proposed change to the [safety injection] SI [recirculation mode transfer] RMT allowable values, and the changes to the LOCA dose consequences analyses. Based on this discussion, the proposed amendments do not increase the probability or consequence of an accident previously evaluated.

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

Response: No.

The proposed change alters the RS pump circuitry by initiating the start sequence with a new RWST Level Low signal instead of a timer after the High High containment pressure setpoint is reached. The timers for the inside RS pumps will be used to sequence pump starts and preclude diesel generator overloading. The RS pump function is not changed. The RWST Level Low instrumentation will be included as part of the Engineered Safety Features Actuation System (ESFAS) instrumentation in the North Anna TS and will be subject to the ESFAS surveillance requirements. The design of the RWST Level Low instrumentation complies with all applicable regulatory requirements and design criteria. The failure modes have been analyzed to ensure that the RWST Level Low circuitry can withstand a single active failure without affecting the RS system design functions. The RS system is an accident mitigation system only, so no new accident initiators are created.

The remaining changes to the containment analysis methodology, the containment air partial pressures, the maximum containment temperature operating limit, the TS allowable values for SI RMT, and the LOCA [alternate source term] AST analysis basis do not impact plant equipment design or function. Together, the changes assure that there is adequate margin available to meet the safety analysis criteria and that dose consequences are within regulatory limits. The proposed changes do not introduce failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously identified.

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

Response: No.

The changes to the actuation of the RS pumps and the increased containment air partial pressure have created an adverse effect on the containment response analyses and the LOCA dose analysis. Analyses have been performed that show the containment design basis limits are satisfied and the post-LOCA offsite and control room doses meet the required criteria for the proposed changes to the containment analysis methodology, the RS pump start method, the TS containment air partial pressure limits, the TS containment temperature maximum limit, the TS allowable values for SI RMT, and the LOCA AST bases. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: 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 Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: November 16, 2006.

Description of amendment request: The proposed amendments would add a reference in Technical Specification (TS) 6.2.C, ''Core Operating Limits Report (COLR),'' to permit the use of the Westinghouse Best-Estimate Large Break Loss of Coolant Accident (BE–LBLOCA) analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of LBLOCA.

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

1. The probability of occurrence or the consequences of an accident previously evaluated are not significantly increased.No physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA (BE–LBLOCA) analysis methodology. The proposed TS change simply involves updating the references in TS 6.2.C, Core Operating Limits Report (COLR), to reference the Westinghouse BE–LBLOCA analysis methodology. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, ''Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.'' No other accident consequence is potentially affected by this change.

All systems will continue to be operated in accordance with current design requirements under the new analysis, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Protection System (RPS) or Engineering Safety Features (ESF) setpoints because of the new analysis methodology.

An analysis of the LBLOCA accident for Surry Units 1 and 2 has been performed with the Westinghouse BE–LBLOCA analysis methodology using ASTRUM. The analysis was performed in compliance with all the NRC conditions and limitations as identified in WCAP–16009–P–A. Based on the analysis results, it is concluded that the Surry Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

There are no changes to assumptions of the radiological dose calculations. Hence, there is no increase in the predicted radiological consequences of accidents postulated in the UFSAR.

Therefore, neither the probability of occurrence nor the consequences of an accident previously evaluated is significantly increased.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created.

The use of the Westinghouse BE–LBLOCA analysis methodology with ASTRUM does not impact any of the applicable design criteria and all pertinent licensing basis criteria will continue to be met. Demonstrated adherence to the criteria in 10 CFR 50.46 precludes new challenges to components and systems that could introduce a new type of accident. Safety analysis evaluations have demonstrated that the use of Westinghouse BE–LBLOCA analysis methodology with ASTRUM is acceptable. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the Westinghouse BE–LBLOCA analysis methodology with ASTRUM does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. Furthermore, no changes have been made to any RPS or ESF actuation setpoints. Based on this review, it is concluded that no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced.

It has been shown that the analytical technique used in the Westinghouse BE–LBLOCA analysis methodology using ASTRUM realistically describes the expected behavior of the reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs have been evaluated. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46 continue to be satisfied.

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

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

Attorney for licensee: 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 Branch Chief: Evangelos C. Marinos

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*.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: December 1, 2005.

Brief description of amendment: The amendment revised Technical Specification 3.6.4.1, ''Secondary Containment.'' Specifically, the amendment revised Surveillance Requirement (SR) 3.6.4.1.4 and SR 3.6.4.1.5 to clarify their intent with respect to secondary containment boundary integrity.

Date of issuance: November 17, 2006.

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

Amendment No.: 175.

Facility Operating License No. NPF–62: The amendment revised the Technical Specification Surveillance Requirements and License.

Date of initial notice in ***Federal Register:*** March 28, 2006 (71 FR 15481).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 2006.

No significant hazards consideration comments received: No.

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: September 26, 2006, as supplemented by the letter dated November 3, 2006.

Brief description of amendments: The amendments revise TS 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' to include specific requirements for the MSIV actuator trains.

Date of issuance: November 17, 2006.

Effective date: Effective as of the date of issuance to be implemented within 10 days from the date of issuance.

Amendment Nos.: Unit 1—163, Unit 2—163, Unit 3—163.

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

Date of initial notice in ***Federal Register:*** October 5, 2006 (71 FR 58879). The supplemental letter dated November 3, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 27, 2006.

Brief description of amendments: The amendments revise Technical

Specification 4.2.1, ''Fuel Assemblies,'' to permit up to four lead fuel assemblies (LFAs) with advanced cladding material to be re-inserted into either the Unit 1 or Unit 2 core for the next operating cycle, which is Cycle 19 for Unit 1 and Cycle 17 for Unit 2. Two of these LFAs were manufactured by Westinghouse Electric Company and contain a limited number of fuel rods with advanced zirconium-based alloys. The other two LFAs were manufactured by Framatome ANP, Inc. with fuel rod cladding material classified as M5™ alloy. These LFAs were originally inserted into the Unit 2 core in April 2003 (Operating Cycles 15 and 16) and are scheduled to be discharged during the 2007 refueling outage.

Date of issuance: November 16, 2006.

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

Amendment Nos.: 280 and 257.

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

*Date of initial notice in **Federal Register:*** March 28, 2006 (71 FR 15482).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated November 16, 2006.

No significant hazards consideration comments received: No

Exelon Generation Company, LLC, Docket No. 50–249, Dresden Nuclear Power Station, Unit 3, Grundy County, Illinois

Date of application for amendment: July 21, 2006, as supplemented by letter dated October 19, 2006.

Brief description of amendment: The amendment revised the values of the safety limit minimum critical power ratio in Technical Specification Section 2.1.1, ''Reactor Core SLs [Safety Limits].''

Date of issuance: November 7, 2006.

Effective date: As of the date of issuance and shall be implemented prior to startup for cycle 20.

Amendment Nos.: 213.

Renewed Facility Operating License Nos. DPR–19 and DPR–25: The amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** August 29, 2006 (71 FR 51228). The October 19, 2006 supplement provided additional clarifying information that did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 7, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 21, 2006, as supplemented on September 6 and October 10, 2006.

Brief description of amendment: The amendment changed the Technical Specifications (TSs) to: (1) Revise TS Section 2.3(4) to change the reactor containment building sump buffering agent from trisodium phosphate to sodium tetraborate and change the TS section title to ''Containment Sump Buffering Agent Specification and Volume Requirement,'' (2) revise TS 3.6(2)d to require a volume of sodium tetraborate that is within an area of acceptable operation, as shown in TS Figure 2–3, and (3) an administrative correction to TS 3.6(2)d(i). The amendment allows OPPD to replace the trisodium phosphate in the containment with sodium tetraborate. Changes were also made to the corresponding TS Bases. The TS changes are approved for Cycle 24 only, ending in the spring 2008 refueling outage.

Date of issuance: November 13, 2006.

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

Amendment No.: 247.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 30, 2006 (71 FR 51646). The September 6 and October 10, 2006, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated November 13, 2006.

No significant hazards consideration comments received: No.

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 application for amendments: November 3, 2005, as supplemented by letters dated May 1, August 15, and October 5, 2006.

Brief description of amendments: The amendments revised Technical Specification Section 5.5.2.11 to modify the definitions of steam generator tube ''Repair Limit'' and ''Tube Inspection.'' The changes define the extent of the required tube inspections and repair criteria within the tubesheet regions.

Date of issuance: November 9, 2006.

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

Amendment Nos.: Unit 2—206; Unit 3—198.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72676). The May 1, August 15, and October 5, 2006, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 9, 2006.

No significant hazards consideration comments received: No.

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 application for amendments: July 14, 2006.

Brief description of amendments: The amendments deleted duplicative notifications, reporting, and restart requirements if a safety limit was violated; replaced plant-specific position titles with generic position titles; and additional administrative changes.

Date of issuance: November 15, 2006.

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

Amendment Nos.: Unit 2—207; Unit 3—199.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 12, 2006 (71 FR 53720).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 15, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 11, 2006.

Brief description of amendment: The amendment revised Surveillance Requirements (SRs) 3.7.2.1, 3.7.3.1, and 3.7.3.3 on verifying the closure time of the main steam isolation valves (MSIVs), main feedwater regulating valves (MFRVs), main feedwater regulating valve bypass valves (MFRVBVs), and main feedwater isolation valve (MFIVs) in the Technical Specifications (TS). These valves are the Main Steam and Main Feedwater System isolation valves. The revisions replace (1) the specified maximum acceptable valve closure time for the MSIVs, MFRVs, and MFRVBVs, and (2) TS Figure 3.7.3–1, which shows acceptable valve closure times for the MFIVs, by the reference to the valve closure time is verified to be ''within limits.'' The maximum acceptable valve closure times for the MFRVs and MFRVBVs, and TS Figure 3.7.3–1 are now located in the TS Bases. The maximum acceptable valve closure time for the MSIV is already in the TS Bases.

Date of issuance: November 15, 2006.

Effective date: Effective as of its date of issuance, and shall be implemented within 90 days of the date of issuance.

Amendment No.: 176.

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

Date of initial notice in ***Federal Register:*** June 20, 2006 (71 FR 35461).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 2006.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, 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 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 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).

Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California

Date of application for amendment: October 18, 2006, as supplemented on November 2, 2006.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 3.8.4, ''DC Sources—Operating,'' Condition B to extend the completion time (CT) to restore an inoperable vital battery from 2 hours to 4 hours for the current operating Cycle 14, provided certain required actions are taken. The extended CT would allow sufficient time to correct a degraded condition on the station Vital Battery 1–1.

Date of issuance: November 15, 2006

Effective date: As of its date of issuance and shall be implemented within 7 days of the date of issuance.

Amendment No.: 190

Facility Operating License No. DPR–80: The amendment revised the Technical Specifications and license.

Public comments requested as to proposed no significant hazards

[1] To the extent that the applications contain attachments and supporting documents that are not publicly avaialble 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.

consideration (NSHC): Yes. An individual 14-day Notice of Consideration of Issuance of Amendment to Facility Operating License was published on October 27, 2006 (71 FR 63040) in the **Federal Register.** 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 26, 2006, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The November 2, 2006, supplemental letter provided additional information that clarified the application, and did not expand the scope of the application as originally noticed.

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 15, 2006.

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

NRC Branch Chief: David Terao

Dated at Rockville, Maryland, this 22nd day of November 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–20329 Filed 12–4–06; 8:45 am]

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NUCLEAR WASTE TECHNICAL REVIEW BOARD

No Fear Act Notice

On May 15, 2002, Congress enacted the ''Notification and Federal Employee Antidiscrimination and Retaliation Act of 2002,'' which is now known as the No FEAR Act. One purpose of the act is to ''require that Federal agencies be accountable for violations of antidiscrimination and whistleblower protection laws'' (Pub. L. 107–174, Summary). In support of this objective, Congress found that ''agencies cannot be run effectively if those agencies practice or tolerate discrimination,'' Public Law 107–174, Title I, General Provisions, section 101(1).

The Act requires the U.S. Nuclear Waste Technical Review Board (Board) to provide this notice to Board employees, former Board employees, and applicants for Board employment to inform them of their rights and protections under Federal antidiscrimination and whistleblower protection laws.

Antidiscrimination Laws/Bases for Complaints or Grievances

The Board cannot discriminate on the basis of race, color, religion, sex, national origin, age, disability, marital status, or political affiliation against an employee or applicant for employment related to the terms, conditions, or privileges of employment. Discrimination on these bases is prohibited by one or more of the following statutes: 5 U.S.C. 2302(b)(1); 29 U.S.C. 206(d); 29 U.S.C. 631; 29 U.S.C. 633a; 29 U.S.C. 791; and 42 U.S.C. 2000e–16.

If you believe that you have been the victim of unlawful discrimination on the basis of race, color, religion, sex, national origin or disability, you must contact an Equal Employment Opportunity (EEO) counselor at General Services Administration within 45 calendar days of the alleged discriminatory action, or, in the case of a personnel action, within 45 calendar days of the effective date of the action, before filing a formal complaint of discrimination with the Board (*See,* e.g., 29 CFR 1614). If you believe that you have been the victim of unlawful discrimination on the basis of age, you must *either* (1) contact an EEO counselor as noted above or (2) give notice of intent to sue to the Equal Employment Opportunity commission (EEOC) within 180 calendar days of the alleged discriminatory action. If you are alleging discrimination bases on marital status or political affiliation, you may file a written complaint with the U.S. Office of Special Counsel (OSC) (see contact information below). As an alternative (or in some cases, in addition), you may pursue a discrimination complaint by filing a grievance through the Board's Administrative Grievance Procedure or 29 CFR part 1614, if such procedures apply and are available.

Whistleblower Protection Laws

A Board employee with authority to take, direct others to take, recommend or approve any personnel action must not use that authority to take, threaten to take, or fail to take a personnel action against an employee or applicant because of disclosure of information by that individual that is reasonably believed to evidence violations of law, rule, or regulation; gross mismanagement; gross waste of funds; an abuse of authority; or a substantial and specific danger to public health or safety; unless disclosure of such information is specifically prohibited by law and such information is specifically required by Executive Order to be kept secret in the interest of national defense or the conduct of foreign affairs.

Retaliation against an employee or applicant for making a protected disclosure is prohibited by 5 U.S.C. 2302(b)(8). If you believe that you have been the victim of whistleblower retaliation, you may file a written complaint (Form OSC–11) with the U.S. Office of Special Counsel (OSC) at 1730 M Street, NW., Suite 218, Washington, DC 20036–4505 or online through the OSC Web site at *http://www.osc.gov.*

Retaliation for Engaging in Protected Activity

The Board cannot retaliate against an employee or applicant because that individual exercises his or her rights under any of the Federal antidiscrimination or whistleblower protection laws listed above. If you believe that you are the victim of retaliation for engaging in protected activity, you must follow, as appropriate, the procedures described in the Antidiscrimination Laws and Whistleblower Protection Laws or, if applicable, the Board's Administrative Grievance Procedure in order to pursue any legal remedy.

Disciplinary Actions

Under existing laws, the Board retains the right, where appropriate, to discipline an employee for conduct that is inconsistent with Federal Antidiscrimination and Whistleblower Protection Laws up to and including removal. If, however, OSC has initiated an investigation under 5 U.S.C. 1214, according to 5 U.S.C. 1214(f), the Board must seek approval from the Special Counsel to discipline an employee for, among other activities, engaging in prohibited retaliation. Nothing in the No FEAR Act alters existing laws or permits the Board to take unfounded disciplinary action against a Federal employee or to violate the procedural rights of a Federal employee who has been accused of discrimination.

Additional Information

For further information regarding the No FEAR Act regulations, refer to 5 CFR part 724. Additional information regarding Federal antidiscrimination, whistleblower protection and retaliation laws can be found at the EEOC Web site at *http://www.eeoc.gov* and the OSC Web site at *http://www.osc.gov.*