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

Dated: December 15, 2005.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 05–24323 Filed 12–16–05; 2:18 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission 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 23, 2005 to December 8, 2005. The last biweekly notice was published on December 6, 2005 (70 FR 72667).

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

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

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

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

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

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station (OCNGS), Ocean County, New Jersey

Date of amendment request: October 18, 2005.

Description of amendment request: The licensee proposes to revise the OCNGS Technical Specifications Surveillance Requirement 4.4.B.1 to provide an alternative means for testing the electromatic relief valves located on the main steam system. The proposed change would allow demonstration of the capability of the valves to perform their function without requiring that the valves be cycled with steam pressure while installed.

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

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

Response: No.

The proposed change modifies Technical Specifications (TS) Surveillance Requirement (SR) 4.4.B.1 to provide an alternative means for testing the Electromatic Relief Valves (EMRVs). Accidents are initiated by the malfunction of plant equipment, or the failure of plant structures, systems, or components. The performance of EMRV testing is not a precursor to any accident previously evaluated and does not change the manner in which the valves are operated. The proposed testing requirements will not contribute to the failure of the relief valves nor any plant structure, system, or component. AmerGen Energy Company, LLC (AmerGen) has determined that the proposed change in testing methodology provides an equivalent level of assurance that the relief valves are capable of performing their intended safety functions. Thus, the proposed change does not affect the probability of an accident previously evaluated.

The performance of EMRV testing provides confidence that the EMRVs are capable of depressurizing the reactor pressure vessel (RPV). This will protect the reactor vessel from overpressurization and allow the Core Spray system to inject into the RPV as designed. The proposed change involves the manner in which the EMRVs are tested, and has no effect on the types or amounts of radiation released or the predicted offsite doses in the event of an accident. The proposed testing requirements are sufficient to provide confidence that the EMRVs are capable of performing their intended safety functions. In addition, a stuck open EMRV accident is analyzed in the Updated Final Safety Analysis Report (section 15.6.1). Since the proposed testing requirements do not alter the assumptions for the stuck open EMRV accident, the consequences of any accident previously evaluated are not increased.

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

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

Response: No.

The proposed change does not affect the assumed accident performance of the EMRVs, nor any plant structure, system, or component previously evaluated. The proposed change does not involve the installation of new equipment, and installed equipment is not being operated in a new of different manner. The change in test methodology ensures that the EMRVs remain capable of performing their safety functions. No set points are being changed which would alter the dynamic response of plant equipment. Accordingly, no new failure modes are introduced.

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

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

Response: No.

The proposed change will allow testing of the EMRV actuation electrical circuitry, including the solenoid, and mechanical actuation components, without causing the EMRV to open. Accordingly, in-situ EMRV cycling is avoided, reducing the potential for valve seat leakage. The valves will be tested in accordance with the Inservice Test (IST) Program that involves testing the valve at a test facility using steam. The combination of the IST and proposed actuator test provides confidence that the EMRVs will perform their design function.

The proposed change does not affect the EMRV set points or the operational criteria that directs the EMRVs to be manually opened during plant transients. There are no changes proposed which alter the set points at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

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 proposed amendment involves no significant hazards consideration.

Attorney for licensee: Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LCC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: October 27, 2005.

Description of amendment request: This amendment proposes revisions to the Technical Specification (TS) Surveillance Requirements (SR) 4.5.2e (Safety Injection), 4.6.2.1d (Containment Spray), and 4.7.3b (Component Cooling Water/Auxiliary Component Cooling Water), by removing the words ''during shutdown.'' Additionally, a revision to delete TS SR 4.7.12.1c (Essential Services Chilled Water) is requested.

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.

Deletion of TS SR 4.7.12.1c is an administrative change since there are no valves in the essential services chilled water system for which the TS SR 4.7.12.1c is applicable. The deletion of the ''during shutdown'' restriction from TS SRs 4.5.2e (Safety Injection), 4.6.2.1d (Containment Spray), and 4.7.3b (Component Cooling Water/Auxiliary Component Cooling Water) does not impact system operation nor does it reduce TS SRs. Component actuations that will be allowed to be performed online for these TS SRs are either already actuated online for other TS SRs or the components to be actuated online are currently stroked online in accordance with the Inservice Testing Program. Therefore, the accident mitigation features of the plant for previously evaluated accidents are not affected by the proposed amendment.

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.

Deletion of TS SR 4.7.12.1c is an administrative change since there are no valves in the essential services chilled water system for which the TS SR 4.7.12.1c is applicable. The deletion of the ''during shutdown'' restriction from TS SRs 4.5.2e (Safety Injection), 4.6.2.1d (Containment Spray), and 4.7.3b (Component Cooling Water/Auxiliary Component Cooling Water) does not impact system operation nor does it reduce TS SR. Component actuations that will be allowed to be performed online for these TS SRs are either already actuated online for other TS SRs or the components to be actuated online are currently stroked online in accordance with the Inservice Testing Program. Therefore, the proposed change introduces no new mode of plant operation and no new possibility for an accident is introduced.

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.

There are no automatic valves in the essential services chilled water system that actuate on an SIAS [safety injection actuation signal]. Deletion of the ''during shutdown'' limitation does not change the TS test requirements or surveillance frequency. Therefore, existing TS surveillance requirements are not reduced by the proposed change, thus no margins of safety are reduced.

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

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

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

NRC Branch Chief: David Terao

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of amendment request: November 7, 2005

Description of amendment request: The amendment would revise the Technical Specifications (TSs), to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment includes changes to the TS definition of Leakage, TS 3/4.4.6, ''Reactor Coolant System Leakage,'' TS 3/4.4.5, ''Steam Generators,'' and adds TS 6.19, ''Steam Generator (SG) Program,'' and TS 6.9.7, ''Steam Generator Tube Inspection Report.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated November 7, 2005.

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

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

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

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

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

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident.

The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Richard J. Laufer.

Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: October 21, 2005.

Description of amendment request: The proposed amendment will revise the Technical Specifications to allow operation with a reduced reactor coolant system (RCS) flow rate of 300,000 gpm and a reduction in the maximum thermal power to 89 percent of the rated thermal power. The definition of rated thermal power remains unchanged at 2700 MWt. The flow rate of 300,000 gpm is expected to conservatively bound an analyzed steam generator tube plugging level of 42 percent per steam generator. The re-analysis performed to support this reduction in RCS flow used Westinghouse WCAP–9272–P–A methodology, the same methodology approved for St. Lucie Unit 2 in License Amendment 138.

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.

None of the proposed changes to the Technical Specifications results in operation of the facility that adversely affects the initiation of any accident previously evaluated. There is no adverse impact on any plant system. Plant systems will continue to function as designed, and all performance requirements for these systems remain acceptable. The analysis, performed to support the proposed changes, has included evaluations and/or analyses of all the analyzed accident analyses, including the effects of changes on the SG tube sleeve design. The analyses and evaluations have verified that the accident analyses acceptance criteria continue to be met. Dose consequences acceptance criteria have been verified to be met for analyzed events. Therefore, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes to the Technical Specifications. Although the allowable tube plugging level is increased, the criteria for tube plugging/sleeving and the tube integrity considerations remain unchanged. The proposed changes have no adverse effects on any safety-related systems and do not challenge the performance or integrity of any safety-related system. The DNBR [Departure from Nucleate Boiling Ratio] limits and trip setpoints associated with the respective reactor protection system functions have verified that the accident analyses criteria continue to be met. Therefore, this amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The safety analyses of all analyzed design basis accidents, supporting the proposed changes to the Technical Specifications, continue to meet the applicable acceptance criteria with respect to the radiological consequences, specified acceptable fuel design limits (SAFDLs), primary and secondary overpressurization, and 10 CFR 50.46 requirements. The DNBR and the setpoint analyses are performed on a cycle-specific basis to verify that the reactor protection system functions continue to provide adequate protection against fuel design limits. Evaluation of the steam line break and LOCA [Loss of Coolant Accident] mass and energy releases determined that the overall containment response remains acceptable. The performance requirements for all systems have been verified to be acceptable from design basis accidents' consideration. The proposed amendment, therefore, will 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: July 25, 2005.

Description of amendment request: The proposed amendment would add new Technical Specifications requirements to provide limiting conditions for operation (LCOs) and action statements and corresponding surveillance requirements for the Emergency Service Water (ESW) system. In the absence of such new requirement, the current requirement at Section 3.5.A.4 simply specifies that the unit be shutdown within 24 hours.

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

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

Response: No.

The Emergency Service Water (ESW) System is not an accident initiator. The proposed change provides operability requirements and surveillance requirements to ensure the ESW System is operable as required for accident mitigation. The proposed operability requirements and allowed outage time is consistent with the requirements for the systems supported by the ESW System. The [calculated radiological] dose to the public and the Control Room operators [due to a postulated accident] are unaffected by the proposed change. The proposed LCO provides direction with respect to actions to be taken when support systems are inoperable.

The proposed Technical Specifications does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not significantly increase the probability or the consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not introduce new equipment operating modes, nor do they alter existing system relationships. The proposed changes do not introduce new failure modes. They do not alter the equipment required for accident mitigation and they appropriately consider the effects on supported systems when a support system is inoperable. When support systems are inoperable, actions are specified consistent with safe plant operation.

Therefore, the proposed changes will 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 the margin of safety?

Response: No.

The proposed change provides specifications for the ESW System that are consistent with current Technical Specification requirements for other equipment. The proposed changes ensure that the ESW and other support systems will be available when required and provides adequate alternative actions when the support systems are not available. The allowed outage times for the ESW subsystem is consistent with that allowed for other equipment required for accident mitigation. Therefore, the proposed changes do 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: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: October 31, 2005.

Description of amendment request: Omaha Public Power District (the licensee) has proposed to revise the Updated Safety Analysis Report (USAR) Safety Analysis, General, Section 14.1, as well as the radiological consequences analyses for the events of Seized Rotor (SR), Section 14.6.2.8; Main Steam Line Break (MSLB), Section 14.12.6; Control Element Assembly Ejection (CEAE), Section 14.13.4; and Steam Generator Tube Rupture (SGTR), Section 14.14.3. The USAR sections for radiological consequences of events need to be revised because of the planned replacement of the steam generators and pressurizer in the fall of 2006.

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 changes to the USAR discuss the changes to the Seized Rotor (SR), Control Element Assembly Ejection (CEAE), Steam Generator Tube Rupture (SGTR) and Main Steam Line Break (MSLB) events resulting from the installation of the replacement steam generators (RSGs) and the replacement pressurizer (RPZR). These changes do not affect an accident initiator previously evaluated in the USAR or the Technical Specifications and will not prevent any safety systems from performing their accident mitigating function as discussed in the USAR or the Technical Specifications.

In all events evaluated, with the exception of the Control Room dose of the MSLB concurrent iodine spike case, there is no margin reduction. The Control Room dose of the MSLB concurrent iodine spike case is increased from 2.5 rem to 4.5 rem. The calculated doses resulting from the proposed changes to USAR Sections 14.1.6, 14.6.2.8, 14.12.6, 14.13.4 and 14.14.3 remain below the regulatory limits set by 10 CFR 50.67.

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

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

Response: No.

The proposed changes are the result of changes in the analysis of the radiological consequences of the SR, CEAE, SGTR and MSLB events of the replacement of the steam generators (SGs) and the pressurizer. The proposed changes do not modify or install any safety related equipment. They do, however, change the licensing basis by using fuel gap fractions from Reference 7.6 in accordance with previously accepted license applications by other licensees and by assuming shorter concurrent iodine spike durations in accordance with Section 2.2 of Appendix E of RG 1.183, since the activity released during the eight-hour spike duration exceeds the available release.

Therefore, these changes do 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 calculated doses resulting from the proposed changes to USAR Sections 14.1.6, 14.6.2.8, 14.12.6, 14.13.4 and 14.14.3 remain below the regulatory limits set by 10 CFR 50.67. In all events evaluated, with the exception of the Control Room dose of the MSLB concurrent iodine spike case, there is no margin reduction. The Control Room dose of the MSLB concurrent iodine spike case is increased from 2.5 rem to 4.5 rem. This margin reduction is primarily due to the significant delay in the reactor coolant reaching 212 F with the RSGs and RPZR (i.e., at 159.2 hours versus the 10.94 hours applicable to the original steam generators). This analysis has conservatively used a spike duration of 4 hours. If the updated analysis took credit for the percentage of defective fuels associated with Technical Specification concentrations when developing the duration of the concurrent iodine spike (i.e., used 0.28% defective fuel versus the conservatively assumed 1% defective fuel used in the analysis), the analysis would have resulted in an estimated spike duration of 2 hours instead of 4 hours and the control room dose would be significantly reduced.

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

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

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

NRC Branch Chief: David Terao.

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

Date of amendment request: November 8, 2005.

Description of amendment request: The proposed amendment would revise the Fort Calhoun Station (FCS) Technical Specifications (TS) to add a new Limiting Condition for Operation 2.8.3(6) and modify Table 3–4, Table 3–5, and Design Features 4.3.1 to address criticality control during spent fuel cask loading operations in the spent fuel pool. This request applies only to spent fuel cask loading in the spent fuel pool and does not affect the licensing basis or invalidate our existing exemption from the criticality monitoring requirements of Title 10, Code of Federal Regulations (CFR) 70.24 for new and spent fuel storage.

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.

These proposed changes affect only operations in the spent fuel pool during spent fuel cask loading operations. Plant power operations and other spent fuel pool operations are not affected. There are no changes to the design or operation of the power plant that could affect system, component or accident functions resulting from these changes.

Fuel loading into the spent fuel casks in the spent fuel pool will not require any significant changes to spent fuel pool structures, systems, or components, nor will their performance requirements be altered. The potential to handle a spent fuel cask was considered in the original design of the plant. Therefore, the response of the plant to previously analyzed Part 50 accidents and related radiological releases will not be adversely impacted, and will bound those postulated during cask loading activities in the cask loading area.

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

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

Response: No.

These proposed changes affect only operations in the spent fuel pool during spent fuel cask loading operations. Plant power operations and other spent fuel pool operations are not affected. No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for mitigation of an event remain capable of fulfilling their intended design function with these changes to the TS.

Fuel handling procedures and associated administrative controls for movement of spent fuel in the spent fuel pool remain applicable and are being appropriately augmented to accommodate spent fuel cask loading operations. Additionally, the soluble boron concentration required to maintain k_{eff} ≤ 0.95 for postulated accidents associated with cask loading operations was also evaluated. The results of the analyses, using a methodology previously approved by the NRC, demonstrate that the amount of soluble boron assumed to be in the pool water during these postulated accidents (800 ppm [part per million]) is much less than the value at which the spent fuel pool is normally maintained (approximately 1900 ppm).

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

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

Response: No.

An NRC-approved methodology was used to perform the criticality analyses that provide the basis to incorporate a boron concentration and a new burnup versus enrichment curve into the plant Technical Specifications to ensure criticality safety margins are maintained during spent fuel cask loading. Spent fuel casks at FCS are loaded in the spent fuel pool in an area adjacent to the spent fuel racks. No physical segregation such as a wall or gate exists between the spent fuel racks and spent fuel cask loading area. The cask loading area floor is approximately two feet lower than the floor on which the spent fuel racks are located. Therefore, the spent fuel pool water flows in and around the spent fuel racks and spent fuel casks being loaded in a common pool. Neutronic coupling between fuel in the spent fuel racks and fuel in the spent fuel cask has been appropriately considered in the criticality analysis, including accident events that postulate mis-loading of a fresh fuel assembly into the cask and dropping a fuel assembly between the spent fuel racks and spent fuel cask during loading.

The normal condition criticality analysis was performed assuming no soluble boron in the spent fuel pool water and credit for fuel burnup. The proposed new Technical Specification requirement to permit only fuel assemblies with the minimum required burnup versus enrichment to be loaded into the spent fuel cask preserves this analysis basis. The accident condition criticality analysis was performed assuming a minimum of 800 ppm boron in the spent fuel pool during cask loading operations. All analyses account for uncertainties at a 95[-] percent probability/95-percent confidence level. The proposed new Technical Specification requirement to maintain a minimum boron concentration of 800 ppm in the spent fuel pool during spent fuel cask loading operations preserves this analysis basis. For defense-in-depth, the spent fuel pool boron concentration is typically maintained at approximately 1900 ppm during normal operations and would not be expected to be reduced during spent fuel cask loading operations.

Therefore, there is no significant reduction in a margin of safety as a result of this change.

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

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

NRC Branch Chief: David Terao.

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

Date of amendment requests: October 19, 2005.

Description of amendment requests: The proposed change allows a delay time for entering a supported system Technical Specification (TS) when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated October 19, 2005.

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

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

The proposed change allows a delay time for entering a supported system 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. The proposed LCO 3.0.8 defines limitations on the use of the provision and includes a requirement for the licensee to assess and manage the risk associated with operation with an inoperable snubber. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: David Terao.

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

Date of amendment requests: October 19, 2005.

Description of amendment requests: The proposed amendments would update the Technical Specification (TS) 5.3, ''Unit Staff Qualifications,'' operator minimum qualification requirements contained in the March 28, 1980, NRC letter to all licensees with the more recent NRC-approved operator qualification requirements contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.1–1993. In addition, the proposed changes remove the TS 5.3.1 plant staff retraining and replacement training program requirements which have been superseded by requirements contained in section 50.120 of Title 10 of the Code of Federal Regulations (10 CFR).

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 is an administrative change to revise the Technical Specification (TS) 5.3.1 licensed operator minimum qualification requirements and remove the plant staff retraining and replacement training program requirements from the TS. The proposed change does not directly impact accidents previously evaluated. The Diablo Canyon Power Plant (DCPP) licensed operator training program is accredited by the National Academy for Nuclear Training (NANT) and is based on a systems approach to training consistent with the requirements of 10 CFR 55. Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. The DCPP plant staff retraining and replacement training program meets the requirements of 10 CFR 50.120.

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 accident from any accident previously evaluated?

Response: No.

The proposed change is administrative in nature and does not affect the plant design, hardware, system operation, or operating procedures. The DCPP licensed operator training program is accredited by the NANT and is based on a systems approach to training consistent with the requirements of 10 CFR 55. Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. The DCPP plant staff retraining and replacement training program meets the requirements of 10 CFR 50.120.

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

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

Response: No.

The proposed change is administrative in nature and does not affect the plant design, hardware, system operation, or operating procedures. The change does not exceed or alter a design basis or safety limit and thus does not reduce the margin of safety.

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

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

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.

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 19, 2005.

Description of amendment request: The amendment would relocate the Technical Specification response time testing tables to the Updated Final Safety Analysis Report.

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

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

Response: No.

The proposed amendment[s] relocate the instrument response time limits for the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) from the technical specifications to the Updated Final Safety Analysis Report (UFSAR). The proposed amendment[s] conform to the guidance given in Enclosures 1 and 2 of Generic Letter 93–08. Neither the response time limits nor the surveillance requirements for performing response time testing will be altered by this submittal. The overall RTS and ESFAS functional capabilities will not be changed and assurance that action requirements of the reactor trip and engineered safety features systems are completed within the time limits assumed in the accident analyses is unaffected by the proposed amendment[s].

Therefore, operation of the facility in accordance with the proposed amendment[s] will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

Response: No.

The proposed amendment[s] will not change the physical plant or the modes of plant operation defined in the operating license[s]. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems.

Therefore, operation of the facility in accordance with the proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The measurement of instrumentation response times at the frequencies specified in the technical specification provides assurance that actions associated with the reactor trip and engineered safety features systems are accomplished within the time limits assumed in the accident analyses. The response time limits and the measurement frequencies remain unchanged by the proposed amendment[s].

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Darrell J. Roberts.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: November 15, 2005.

Description of amendment request: The amendment would revise the Virgil

C. Summer Nuclear Station Technical Specifications (TS) 3/4.3.1, ''Reactor Trip System Instrumentation,'' and TS 3/4.3.2, ''Engineered Safety Feature Actuation System Instrumentation,'' to implement the allowed outage time and bypass test time changes approved by the Nuclear Regulatory Commission in the Westinghouse topical report WCAP–14333–P–A, Rev. 1, ''Probabilistic Risk Analysis of the Reactor Trip System and Engineered Safety Features Actuation System Test Times and Completion Times,'' dated October 1998.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the Technical Specifications do not result in a condition where the design, material, and construction standards that were applicable prior to the changes are altered.

The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident. There will be no changes to normal plant operating parameters or accident mitigation performance. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [final safety analysis report]. The determination that the results of the proposed changes are acceptable was established in the NRC SE [safety evaluation] issued for WCAP [Westinghouse Commercial Atomic Power report]-14333, dated July 15, 1998. Implementation of the proposed changes will result in an insignificant risk impact. The proposed changes to Action 16 of TS [Technical Specification] 3/4.3.2 are also acceptable as demonstrated by meeting the acceptance criteria contained in Regulatory Guides 1.174 and 1.177.

The proposed changes to the AOTs [allowable outage times] and bypass test times, reduce the potential for inadvertent reactor trips and spurious ESF [engineered safety feature] actuations, and therefore do not increase the probability of any accident previously evaluated. The proposed changes do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the increase in CDF [core damage frequency] is less than 1.0E–06 per year and the increase in LERF [large early release frequency] is less than 1.0E–07 per year. In addition, for the AOT and bypass test time changes, the ICCDP [incremental conditional core damage probability] and ICLERP [incremental conditional large early release probability] values are less than 5.0E–07 and 5.0E–08, respectively. The proposed changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by the ''CDF, ''LERF, ICCDP, ICLERP risk metrics is within the acceptance criteria of Regulatory Guides 1.174 and 1.177, there will not be a significant increase in the consequences of any accidents.

The proposed changes to the bypass test times and AOTs do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event to within the applicable acceptance criteria. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

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

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

Response: No.

There are no hardware changes or any changes in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in a physical alteration to any plant system or a change in the method by which any safety-related plant system performs its safety function. There will be no setpoint changes or changes to accident analysis assumptions.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

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

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

Response: No.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the DNBR [departure from nucleate boiling ratio] limits, FQ, FDH, LOCA [loss-of-coolant accident] PCT [peak cladding temperature], peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria continue to be met.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased AOTs and bypass test times, it is expected that there would be a net benefit due to the reduced potential for spurious reactor trips and actuations associated with testing and maintenance activities.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to RTS and ESFAS instrumentation Actions with short AOTs.

The increased AOTs will provide more time for trouble shooting and repair activities, therefore reducing the potential for spurious trips and actuations.

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

Pursuant to 10CFR50.91, the preceding analyses provide a determination that the proposed Technical Specification changes pose no significant hazard as delineated by 10CFR50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.929(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: November 2, 2005.

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a Technical Specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions are revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement 3.0.4 is revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated November 2, 2005.

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

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. 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). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, 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 a Margin of Safety

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

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*.

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: January 27, 2005, as supplemented on November 2, 2005.

Brief description of amendments: The amendments modify Technical Specifications (TSs) requirements to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, ''Increased Flexibility in Mode Restraints.''

Date of issuance: December 2, 2005

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

Amendment Nos.: 276 and 253

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

*Date of initial notice in **Federal Register:*** May 10, 2005 (70 FR 24648)

The supplemental letter dated November 2, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated December 2, 2005.

No significant hazards consideration comments received: No

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

Date of application for amendments: November 16, 2004, as supplemented by letters dated May 3, July 6, September 13, October 6, October 24 and November 15, 2005

Brief description of amendments: The amendments revised the Technical Specifications, on a one-time basis, to allow the nuclear service water system headers for each unit to be taken out of service for up to 14 days each for system upgrades.

Date of issuance: November 17, 2005

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

Amendment Nos.: 228/223 *Renewed Facility Operating License Nos. NPF–35 and NPF–52:* Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21454)

The supplements dated May 3, July 6, September 13, October 6, October 24, and November 15, 2005, provided additional information that clarified the application, did not expand the scope of the November 16, 2004 application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 2005.

No significant hazards consideration comments received: No

Duke Energy Corporation, Docket Nos. 50–269 and 50–270, Oconee Nuclear Station, Units 1 and 2, Oconee County, South Carolina

Date of application of amendments: August 18, 2005, as supplemented by letter dated September 15, 2005

Brief description of amendments: The amendments revised the Technical Specifications 3.5.2.6 and 3.5.3.6 to accommodate the replacement of the reactor building emergency sump suction inlet trash racks and screens with strainers.

Date of Issuance: November 1, 2005

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

Amendment Nos.: 348/350

Renewed Facility Operating License Nos. DPR–38 and DPR–47: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 31, 2005 (70 FR 51852)

The supplement dated September 15, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 1, 2005.

No significant hazards consideration comments received: No

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

Date of amendment request: March 8, 2005, as supplemented by letters dated April 19, July 12, September 21, November 14, and November 15, 2005

Brief description of amendment: The amendment enables the licensee to make changes to the Updated Safety Analysis Report (USAR) to reflect the use of the non-single-failure-proof Fuel Building Cask Handling Crane for dry spent fuel cask component lifting and handling operations.

Date of issuance: December 1, 2005

Effective date: As of the date of issuance, with the implementation to begin immediately and be completed by the next periodic update to the USAR, in accordance with 10 CFR 50.71(e).

Amendment No.: 149

Facility Operating License No. NPF–47: The amendment allows revision of the USAR.

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21455). The supplemental letters dated April 19, July 12, September 21, November 14, and November 15, 2005, 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 December 1, 2005.

No significant hazards consideration comments received: No

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

Date of amendment request: May 25, 2005

Brief description of amendment: The amendment deleted from the Cooper Nuclear Station Technical Specifications temporary footnotes that have expired and are no longer in effect.

Date of issuance: December 5, 2005

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

Amendment No.: 213

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 5, 2005 (70 FR 38721)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 5, 2005.

No significant hazards consideration comments received: No

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

Date of application for amendments: January 28, 2005

Brief description of amendments: The amendments replace the reference to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with a reference to ASME Code for Operation and Maintenance of Nuclear Power Plants in Technical Specification 5.5.6.

Date of issuance: December 7, 2005

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

Amendment Nos.: 228 and 204

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29799)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 2005.

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: September 9, 2005, as supplemented by letters dated October 24 and November 3, 2005

Brief description of amendment: The amendment revises Surveillance Requirements (SRs) 3.7.3.1 and 3.7.3.2 and adds SR 3.7.3.3 in TS 3.7.3, ''Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and Main Feedwater Regulating Valve Bypass Valves (MFRVBVs).'' The amendment also adds Figure 3.7.3–1 to the TSs to specify the acceptable MFIV stroke, or closure, time with respect to steam generator pressure.

Date of issuance: November 17, 2005

Effective date: Effective as of its date of issuance, and shall be implemented no later than entry into Mode 3 during the startup from Refueling Outage 15, which is scheduled for the spring of 2007. Completion of the baseline testing of the main feedwater isolation valves, which is described in the licensee's letters dated September 9 and October 24, 2005, and in Section 4.1.4 of the Safety Evaluation for this amendment, shall be completed as part of the implementation of this amendment.

Amendment No.: 170

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

Date of initial notice in ***Federal Register:*** September 16, 2005 (70 FR 54776)The supplemental letters dated October 24 and November 3, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 2005.

No significant hazards consideration comments received: No

Dated at Rockville, Maryland, this 12th day of December 2005.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. 05–24142 Filed 12–19–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Privacy Act of 1974, as Amended; Revisions to Existing System of Records

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed revisions to an existing system of records.

SUMMARY: The Nuclear Regulatory Commission (NRC) is issuing public notice of its intent to modify an existing system of records, NRC–20, ''Official Travel Records—NRC,'' to incorporate the collection and use of travel charge card records, including credit data, to comply with the Consolidated Appropriations Act, 2005 (Pub. L. 108–447).

DATES: The revised system of records will become effective without further notice on January 30, 2006 unless comments received on or before that date cause a contrary decision. If changes are made based on NRC's review of comments received, a new final notice will be published.

ADDRESSES: Comments may be provided to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Written comments should also be transmitted to the Chief of the Rules and Directives Branch, either by means of facsimile transmission to (301) 415–5144, or by e-mail to *nrcrep@nrc.gov.*

FOR FURTHER INFORMATION CONTACT: Sandra S. Northern, Privacy Program Officer, FOIA/Privacy Act Team, Records and FOIA/Privacy Services Branch, Information and Records Services Division, Office of Information Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–6879; e-mail: *ssn@nrc.gov.*

SUPPLEMENTARY INFORMATION: NRC is proposing to add new categories of records in the system to include charge card applications, terms and conditions for use of charge cards, charge card training documentation, monthly reports regarding accounts, credit data, and related documentation; update the authority for the system by adding Section 639 of the Consolidated Appropriations Act, 2005 (Pub.L. 108–447); and incorporate three new routine uses which will allow disclosure of information to the charge card issuing bank, the Department of Interior, National Business Center, to collect severe travel card delinquencies by employee salary offset, and to a consumer reporting agency to obtain credit reports.

A report on the proposed revisions is being sent to OMB, the Committee on Homeland Security and Governmental Affairs of the U.S. Senate, and the Committee on Government Reform of the U.S. House of Representatives as required by the Privacy Act and OMB Circular No. A–130, Appendix I, ''Federal Agency Responsibilities for Maintaining Records About Individuals.'' NRC's actions are also consistent with OMB Circular A–123, ''Management's Responsibility for Internal Control.''

Accordingly, the NRC proposes to amend NRC–20 to read as follows:

NRC–20

SYSTEM NAME:

Official Travel Records—NRC.

SYSTEM LOCATION:

Primary system—Division of Financial Services, Office of the Chief Financial Officer, NRC, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland.

Duplicate system—Duplicate systems may exist, in part, within the organization where the employee actually works for administrative purposes, at the locations listed in Addendum I, Parts 1 and 2, published on September 24, 2004 (69 FR 57579).

CATEGORIES OF INDIVIDUALS COVERED BY THE SYSTEM:

Current and former NRC employees, prospective NRC employees, consultants, and invitational travelers for NRC programs.

CATEGORIES OF RECORDS IN THE SYSTEM:

These records contain requests and authorizations for official travel, travel