NATIONAL FOUNDATION ON THE ARTS AND THE HUMANITIES

National Endowment for the Arts; Proposed Collection; Comments Request

SUMMARY: The National Endowment for the Arts (NEA), as part of its continuing effort to reduce paperwork and respondent burden, conducts a preclearance consultation program to provide the general public and federal agencies with an opportunity to comment on proposed and/or continuing collections of information in accordance with the Paperwork Reduction Act of 1995 (PRA95) [44 U.S.C. 3506(c)(A)]. This program helps to ensure that requested data can be provided in the desired format, reporting burden (time and financial resources) is minimized, collection instruments are clearly understood, and the impact of collection requirements on respondents can be properly assessed. Currently, the NEA is soliciting comments concerning the proposed information collection of: National Endowment for the Arts Panelist Profile Form. A copy of the current information collection request can be obtained by contacting the office listed below in the address section of this notice.

DATES: Written comments must be submitted to the office listed in the address section below on or before May 10, 2006. The NEA is particularly interested in comments which:

• Evaluate whether the proposed collection of information is necessary for the proper performance of the functions of the agency, including whether the information will have practical utility;

• Evaluate the accuracy of the agency's estimate of the burden of the proposed collection of information including the validity of the methodology and assumptions used;

• Enhance the quality, utility, and clarity of the information to be collected; and

• Minimize the burden of the collection of information on those who are to respond.

ADDRESSES: Kathy Plowitz-Worden, National Endowment for the Arts, 1100 Pennsylvania Avenue, NW., Room 710, Washington, DC 20506–0001, telephone (202) 682–5421 (this is not a toll-free number), fax (202) 682–5049.

Dated: March 8, 2006.

Murray Welsh,

Director Administrative Services, National Endowment for the Arts.

[FR Doc. E6–3541 Filed 3–13–06; 8:45 am]

BILLING CODE 7537–01–P

NATIONAL TRANSPORTATION SAFETY BOARD

Sunshine Act; Meeting

TIME AND PLACE: 9:30 a.m., Thursday, March 23, 2006.

PLACE: NTSB Conference Center, 429 L'Enfant Plaza SW., Washington, DC 20594.

STATUS: The one item is open to the public.

MATTER TO BE CONSIDERED: 7680B, *Railroad Accident Report*—Collision Between Two Washington Metropolitan Area Transit Authority Trains at the Woodley Park-Zoo/Adams Morgan Station in Washington, DC, November 3, 2004.

News Media Contact: Telephone: (202) 314–6100.

Individuals requesting specific accommodations should contact Chris Bisett at (202) 314–6305 by Friday, March 17, 2006.

The public may view the meeting via a live or archived Webcast by accessing a link under ''News & Events'' on the NTSB home page at *http://www.ntsb.gov.*

FOR MORE INFORMATION CONTACT: Vicky D'Onofrio, (202) 314–6410.

Dated: March 9, 2006.

Vicky D'Onofrio,

Federal Register Liaison Officer.

[FR Doc. 06–2514 Filed 3–10–06; 2:05 pm]

BILLING CODE 7533–01–M

NUCLEAR REGULATORY COMMISSION

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

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 February 16, 2006 to March 2, 2006. The last biweekly notice was published on February 28, 2006 (71 FR 10071).

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.*

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

Date of amendments request: February 14, 2006.

Description of amendments request: The amendments would revise Technical Specifications (TS) 3.6.3 to allow a blind flange to be used for containment isolation in each of the two flow paths of the 42 inch refueling purge valves in Modes 1 through 4 without remaining in TS 3.6.3 Condition D.

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 probability of an accident previously evaluated would not be affected by the proposed changes to allow the use of blind flanges for containment isolation in each of the two 42 inch refueling purge valve flow paths. The blind flanges are passive components that could not initiate an accident.

The consequences of an accident previously evaluated would not be increased because the blind flanges would provide containment isolation assumed in the accident analyses instead of the 42 inch refueling purge valves. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program that is required by TS surveillance requirement (SR) 3.6.1.1 and TS 5.5.16. The blind flanges are sealed using two separate concentric O-rings and are leak rate tested after installation by pressurizing the space between the O-rings through a test connection and measuring the leakage. In addition, the outboard 42 inch refueling purge valve packing leakage is measured by pressurizing the stuffing box through the leak off line after flange installation and after any maintenance on the packing. The sum of the individual leakage rates is compared to the acceptance criteria. The blind flanges are verified to be in position at a frequency of 31 days in accordance with TS SR 3.6.3.3.

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.

A new or different kind of accident from any accident previously evaluated would not be created by the proposed changes to allow the use of blind flanges for containment isolation in each of the two 42 inch refueling purge valve flow paths. The blind flanges are passive components that could not create an accident.

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

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

Response: No.

No margin of safety is affected by the proposed changes to allow the use of blind flanges for containment isolation in each of the two 42 inch refueling purge valve flow paths. The blind flanges would provide containment isolation assumed in the accident analyses instead of the 42 inch refueling purge valves. The blind flanges are passive devices not susceptible to an active failure or malfunction that could result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. The blind flanges are leak rate tested in accordance with the containment leakage rate testing program that is required by TS SR 3.6.1.1 and TS 5.5.16. The blind flanges are leak rate tested after installation by pressurizing the space between the O-rings through a test connection and measuring the leakage. In addition, the outboard 42 inch refueling purge valve packing leakage is measured by pressurizing the stuffing box through the leak off line after flange installation and after any maintenance on the packing. The sum of the individual leakage rates is compared to the acceptance criteria. The blind flanges are verified to be in position at a frequency of 31 days in accordance with SR 3.6.3.3.

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

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Branch Chief: David Terao.

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

Date of amendment request: January 31, 2006.

Description of amendment request: The proposed amendment would address an inconsistency that was inadvertently introduced during conversion to improved technical specifications (TSs) when ''1 per room'' replaced ''2'' as the required channels per trip system for the reactor water cleanup (RWCU) area ventilation differential temperature—high isolation function.

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

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

Response: No.

The proposed change clarifies the requirement to maintain isolation capability for the RWCU Area Ventilation Differential Temperature—High isolation instrumentation by addition of a note to TS 3.3.6.1 Condition B, modification of TS 3.3.6.1 Surveillance Requirements Notes, and by clarifying the number of instruments required to be available in TS Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' Function 5.c, by the addition of note (d). This ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. No changes in operating practices or physical plant equipment are created as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change clarifies the requirement to maintain isolation capability for the RWCU Area Ventilation Differential Temperature—High isolation instrumentation by addition of a note to TS 3.3.6.1 Condition B, modification of TS 3.3.6.1 Surveillance Requirements Notes, and by clarifying the number of instruments required to be available in TS Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' Function 5.c, by the addition of note (d). This ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. No physical change in plant equipment will result from this proposed change. Therefore, the proposed change does not create the possibility of a new or different type of accident from any previously evaluated.

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

Response: No.

The proposed change clarifies the requirement to maintain isolation capability for the RWCU Area Ventilation Differential Temperature—High isolation instrumentation by addition of a note to TS 3.3.6.1 Condition B, modification of TS 3.3.6.1 Surveillance Requirements Notes, and by clarifying the number of instruments required to be available in TS Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' Function 5.c, by the addition of note (d). This ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Timothy J. Kobetz, Acting.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: January 30, 2006.

Description of amendment request: The license amendment request would modify the currently approved radiological accident analyses (RAA) and associated Technical Specifications (TS) to account for the difference between the control room emergency zone (CREZ) unfiltered in-leakage (UFI) assumed in the current RAA and the CREZ UFI that was measured during testing.

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

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

No. There are no system, structural, or component (SSC) alterations due to these changes. The radiological accident analyses inputs modified by this request are not accident initiators and do not affect the frequency of occurrence of previously analyzed transients.

The radiological accident analyses have demonstrated acceptable results using the revised inputs for all affected accidents. Further, the proposed changes do not alter or prevent the ability of structures, systems or components to perform their intended function to mitigate the consequences of accidents previously evaluated in the Updated Safety Analysis Report.

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

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

No. There are no physical changes to the plant SSCs and there is no adverse impact on component or system interactions due to the proposed changes. The modes of operation of the plant remain unchanged and the design functions of all the safety systems remain in compliance with the applicable safety analysis acceptance criteria. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The radiological accident analysis inputs modified by this request were incorporated into the revised radiological accident analyses. The revised radiological analyses satisfy all applicable acceptance criteria. There is no adverse effect on plant safety due to this proposed license amendment. Therefore, the change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

Acting NRC Branch Chief: T. Kobetz.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of amendment request: February 6, 2006.

Description of amendment request: The proposed amendment adds a license condition to extend certain Technical Specification (TS) surveillance test intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005.

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

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

Response: No.

The requested action is a one-time extension to the performance interval of a limited number of TS surveillance requirements. The performance of these surveillances, or the failure to perform these surveillances, is not a precursor to an accident. Performing these surveillances or failing to perform these surveillances does not affect the probability of an accident. Therefore, the proposed delay in performance of the surveillance requirements in this amendment request does not increase the probability of an accident previously evaluated.

A delay in performing these surveillances does not result in a system being unable to perform its required function. In the case of this one-time extension request, the relatively short period of additional time that the systems and components will be in service before the next performance of the surveillance will not affect the ability of those systems to operate as designed. Therefore, the systems required to mitigate accidents will remain capable of performing their required function. No new failure modes have been introduced because of this action and the consequences remain consistent with previously evaluated accidents. Therefore, the proposed delay in performance of the surveillance requirements in this amendment request does not involve a significant increase in the consequences of an accident.

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

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

Response: No.

The proposed amendment does not involve a physical alteration of any system, structure, or component (SSC) or a change in the way any SSC is operated. The proposed amendment does not involve operation of any SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the one-time surveillance requirement deferrals being requested.

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

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

Response: No.

The proposed amendment is a one-time extension of the performance interval of a limited number of TS surveillance requirements. Extending these surveillance requirements does not involve a modification of any TS Limiting Conditions for Operation. Extending these surveillance requirements does not involve a change to any limit on accident consequences specified in the license or regulations. Extending these surveillance requirements does not involve a change to how accidents are mitigated or a significant increase in the consequences of an accident. Extending these surveillance requirements does not involve a change in a methodology used to evaluate consequences of an accident. Extending these surveillance requirements does not involve a change in any operating procedure or process.

The instrumentation and components involved in this request have exhibited reliable operation based on the results of the most recent performance of their 18-month surveillance requirements.

Based on the limited additional period of time that the systems and components will be in service before the surveillances are next performed, as well as the operating experience that these surveillances are typically successful when performed, it is reasonable to conclude that the margins of safety associated with these surveillance requirements will not be affected by the requested extension.

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

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

Acting NRC Branch Chief: T. Kobetz.

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: December 19, 2005.

Description of amendment request: The amendment proposes to revise the Technical Specifications (TS) to make the temporary changes to TS Table 3.3.8.1–1, previously approved by Amendment No. 147, permanent. TS Table 3.3.8.1–1 would be revised to delete the temporary note, correct the number of Required Channels per Division for the Loss of Power (LOP) time delay functions, and delete the requirement to perform Surveillance Requirement (SR) 3.3.8.1.2, the monthly Channel Functional Test, on certain LOP time delay functions.

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 regarding the number of required channels per division for the LOP time delay functions are administrative in nature. The changes do not alter the instrumentation design or their physical configuration, and will not affect their operation or manner of control. The proposed changes correct an inconsistency between a TS Table and the RBS [River Bend Station, Unit 1] design basis. The TS required number of voltage sensors per division and associated channel components that monitor voltage conditions and provide the 4.16 kV bus undervoltage protection are unchanged.

The exclusion of the time delay functions from the monthly Channel Functional Test is proposed because the test creates a loss of function for the LOP instrumentation and is, therefore, undesirable during unit operations. The test also introduces the potential for an unintended plan transient, so the elimination of the requirement reduces the potential for such transients.

The channel functional test will continue to be performed every 31 days for the sensor channels. In addition, the LOP time delay functions will continue to be functionally tested and calibrated every 18 months as required by SR 3.3.8.1.3 and SR 3.3.8.1.4. Therefore, the required LOP instrumentation will continue to be tested in a manner and at a frequency necessary to provide confidence that the instrumentation can perform its intended safety function.

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 changes do not alter the instrumentation design or their physical configuration, and will not affect their operation or manner of control. The proposed TS changes do not introduce any new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

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

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

Response: No.

The proposed changes have no affect on any safety analysis assumptions or methods of performing safety analyses. The changes do not adversely affect system OPERABILITY or design requirements and the equipment continues to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety functions. [Regulation] 10 CFR 50.36(c)(3) requires the TS to include Surveillance Requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The channel functional test will continue to be performed every 31 days for the sensor channels. In addition, the LOP time delay functions will continue to be functionally tested and calibrated every 18 months as required by SR 3.3.8.1.3 and SR 3.3.8.1.4.

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: January 26, 2006.

Description of amendment request: The proposed amendment will modify Technical Specification (TS) requirements to support the implementation of Average Power Range Monitor (APRM), Rod Block Monitor, TS/Maximum Extended Operating Domain (ARTS/MEOD).

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?

The proposed changes revise thermal limit structure employed to comply with TS Section 3.2 LCOs [limiting conditions for operation]. The proposed changes will replace the flow-biased APRM scram and rod block trip setdown requirements with power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) or Linear Heat Generation Rate (LHGR) thermal limits. The adjustments to the thermal limits have been determined using NRC approved analytical methods as required by Technical Specifications 5.6.5.b and topical reports as specified in the Core Operating Limits Report (COLR). The proposed changes will not affect any accident initiating mechanism. Adjustments to thermal limits will be determined using NRC approved methodologies. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any anticipated operational occurrence (AOO), that the fuel thermal and mechanical design bases will be maintained, and that the consequences of the postulated loss of coolant accident (LOCA) will remain within acceptable limits. There are no changes to radioactive source terms or release pathways. Operation within the expanded operating domain has been evaluated and the affect on plant accidents was found to be within acceptable parameters. The proposed changes do not result in any significant change in the availability of logic systems or safety-related systems themselves. Required protective functions will be maintained. The proposed changes do not degrade plant design, operation, or the performance of any safety system assumed to function in the accident analysis.

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?

The proposed changes do not introduce any new accident initiators or failure mechanisms because the changes and the affects on existing structures, systems and components have been evaluated and found to not have any adverse affects. The proposed changes eliminate the requirement for setdown of the flow-biased APRM scram and rod block trip setpoints or APRM adjustments under specified conditions and will substitute adjustments to the MCPR and MAPLHGR or LHGR thermal limits. Because the thermal limits will continue to be met, no transient event will escalate into a new or different type of accident due to the initial starting conditions permitted by the adjusted thermal limits.

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

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There is no affect on the conclusions of any safety analysis. Replacement of the APRM setpoint requirement with power and flow dependent adjustments to the MCPR and MAPLHGR or LHGR thermal limits will continue to ensure that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. The fuel cladding safety limit will not be violated as a result of any anticipated operational occurrence. The flow and power dependent adjustments will be determined using NRC approved methodologies. The flow and power dependent adjustments will also ensure that all fuel thermal-mechanical design bases shall remain within the licensing limits. The proposed changes do not involve any increase in calculated off-site dose consequences. Operability of protective instrumentation and the associated systems is assured, and performance of equipment will not be significantly affected.

Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: January 26, 2006.

Description of amendment request: The proposed license amendment replaces the existing Reactor Vessel Material Surveillance Program with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

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

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

Response: No.

The proposed change to the licensing basis continues to assure that applicable regulatory requirements are met and the same assurance of reactor pressure vessel integrity continues to be provided. The proposed change to the License and licensing basis follow the NRC Safety Evaluation approving the implementation of the ISP. The proposed change ensures that the reactor pressure vessel will continue to be operated within the design, operational, and testing limits.

The proposed change does not modify the reactor coolant pressure boundary, (*i.e.*, there are no changes in operating pressure, materials, or seismic loading). The proposed change does not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected.

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

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

Response: No.

The proposed change does not involve a modification to the design of plant structures, systems, or components. Thus, no new modes of operation are introduced by the proposed change. The proposed change will not create any failure mode not bounded by previously evaluated accidents.

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

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

Response: No.

The proposed implementation of ISP has been previously approved by the NRC and found to provide an acceptable alternative to plant-specific reactor vessel material surveillance programs. Operation of JAFNPP within the program ensures that the reactor vessel materials will continue to behave in a non-brittle manner, thereby preserving the original safety design bases.

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

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: February 15, 2006.

Description of amendment request: The proposed change will specifically credit the measurement tank weir flow instrumentation for the containment fan cooler condensate flow monitoring system in place of the one containment fan cooler condensate flow switch currently required by Technical Specification 3.4.5.1, ''Reactor Coolant System Leakage—Leakage Detection Instrumentation.''

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

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

Response: No.

The Reactor Coolant System (RCS) leakage detections systems are passive monitoring systems; therefore, the proposed changes do not affect reactor operations or accident analyses and have no radiological consequences. The change maintains conservative restrictions on RCS leakage detections systems consistent with Regulatory Guide 1.45 [''Reactor Coolant Pressure Boundary Leakage Detection Systems''] and 10 CFR [Part] 50, Appendix A, General Design Criteri[on] 30.

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 introduces no new mode of plant operation or any plant modification. The RCS leakage detection instrumentation is not part of plant control instruments or engineered safety feature actuation circuits but is used solely for monitoring purposes. The change does not vary or affect any plant operating condition or parameter.

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 will be no adverse affects on margins of safety since more stringent requirements will be applied to the third method (CFC [Containment Fan Cooler] condensate flow monitoring) of detecting RCS leakage. The third required RCS leakage detection method will now be capable of detecting a one gallon per minute leak within one hour.

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.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: October 3, 2005.

Description of amendment request: The proposed amendments would revise the reactor coolant system pressure and temperature limits report (PTLR) requirements. Specifically, the amendment would revise the TS Section 1.1, ''Definitions,'' description of the PTLR by deleting reference to specifications containing limits in the PTLR; (2) revise the administrative controls TS 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' by requiring the NRC approval documents to be identified by date and topical reports to be identified by number and title in accordance with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–419; ''Revise PTLR Definition and References in ISTS 5.6.6, RC PTLR,'' and (3) add Westinghouse Electric Company, LLC, WCAP–16143, ''Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Byron/Braidwood Units 1 and 2,'' to the list of analytical methods provided in TS 5.6.6.

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

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

Response: No.

The proposed change to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P–T [pressure temperature] limits.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine the pressure temperature (P–T) limits or Low Temperature Overpressure Protection (LTOP) System setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5.6.6.b requires that the analytical methods used to determine the P–T limits be those previously reviewed and approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59, ''Changes, tests and experiments,'' and where required receive NRC review and approval.

The use of WCAP–16143, following approval by the NRC, for generation of P–T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

The proposed changes 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 within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Based on the above discussion, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P–T limits.

The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P–T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5.6.6.b requires that the analytical methods used to determine the P–T limits be those previously reviewed and approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The use of WCAP–16143, following approval by the NRC, for generation of P–T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

The proposed changes will allow the use of a new NRC-approved methodology for the calculation of P–T limits. However, the changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) and do not introduce a new mode of plant operation. Safety functions associated with P–T limits and LTOP setpoints will continue to function as previously assumed in accident analyses.

Based on this evaluation, 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 to the definition of PTLR is considered to be an editorial change because the requirements of TS 5.6.6 continue to specify the Limiting Conditions for Operation that address operation within the P–T limits. The proposed changes to reference only the Topical Report Number and title do not alter the use of the analytical methods used to determine the P–T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license provided there is no change to the approved methodology. TS 5 .6.6.b requires that the analytical methods used to determine the P–T limits be those previously reviewed and approved by the NRC. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

The P–T limits provide assurance that the reactor pressure vessel is maintained. The use of WCAP–16143, following approval by the NRC, for generation of P–T limits will continue to ensure that reactor pressure vessel integrity is maintained under all conditions.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. Changes to setpoints at which protective actions are initiated that are allowed by the use of WCAP–16143 are evaluated in accordance with 10 CFR 50.59 and where required receive NRC review and approval. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

Based on this evaluation, 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. J. Bradley Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Mindy Landau, Acting.

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: January 25, 2006.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 1.1, ''Definitions,'' and TS 3.4.16, ''RCS Specific Activity.'' The proposed amendments would replace the current TS 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new DOSE EQUIVALENT XE–133 definition (corresponding to the Xenon-133 isotope) that would replace the current—AVERAGE DISINTEGRATION ENERGY definition. In addition, the current DOSE EQUIVALENT I–131 definition (corresponding to the Iodine-131 isotope) would be revised to allow the use of alternate, NRC-approved thyroid dose conversion factors.

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 add a new thyroid dose conversion factor reference to the definition of DOSE EQUIVALENT I–131, eliminate the definition of Ē—AVERAGE DISINTEGRATION ENERGY, add a new definition of DOSE EQUIVALENT XE–133, replace the Technical Specification (TS) 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a limit on noble gas specific activity in the form of a Limiting Condition for Operation (LCO) on DOSE EQUIVALENT XE–133, replace TS Figure 3.4.16–1 with a maximum limit on DOSE EQUIVALENT I–131, extend the Applicability of LCO 3.4.16, and make corresponding changes to TS 3.4.16 to reflect all of the above are not accident initiators and have no impact on the probability of occurrence for any design[-]basis accidents.

The proposed changes will have no impact on the consequences of a design[-basis accident because they will limit the RCS noble gas specific activity to be consistent with the values assumed in the radiological consequence analyses. The changes will also limit the potential RCS iodine concentration excursion to the value currently associated with full power operation, which is more restrictive on plant operation than the existing allowable RCS iodine specific activity at lower power levels.

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 accident from any accident previously evaluated?

Response: No.

The proposed changes do not alter any physical part of the plant nor do they affect any plant operating parameters besides the allowable specific activity in the RCS. The changes that impact the allowable specific activity in the RCS are consistent with the assumptions assumed in the current radiological consequence analyses.

Therefore, the proposed changes do 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 acceptance criteria related to the proposed changes involve the allowable control room and offsite radiological consequences following a design[-]basis accident. The proposed changes will have no impact on the radiological consequences of a design[-]basis accident because they will limit the RCS noble gas specific activity to be consistent with the values assumed in the radiological consequence analyses. The changes will also limit the potential RCS iodine specific activity excursion to the value currently associated with full power operation, which is more restrictive on plant operation than the existing allowable RCS iodine specific activity at lower power levels.

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

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

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

NRC Branch Chief: David Terao.

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: October 28, 2005.

Description of amendment request: The amendment would revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) TS 3.8.1 to incorporate changes implementing requirements for an Alternate AC (AAC) power supply.

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

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

No.

The proposed change revised two action statements and relocated a surveillance requirement. The first AOT [allowable outage time] extension permits one EDG [emergency diesel generator] to be inoperable for up to 14 days, but the AAC [alternate alternating current] source will have to be available. This proposed change will be primarily used for scheduled preventative maintenance while the plant is online. If used for corrective maintenance, the AAC source will have to be capable of providing power within one hour, otherwise the existing 72-hour AOT would apply. This assures that adequate power remains available to the ESF buses to enable the plant to safely shut down, maintain a safe shutdown condition, and/or mitigate the effects of a design basis accident.

The second AOT extension provides an additional two hours to complete the verification of supported equipment for operability. This additional time allows for a planned and systematic approach to performing this verification. Since there are other more immediate ways for the control room staff to be notified of the inoperable status of ESF [engineered safety feature] equipment, (annunciators, BISI, status lights), the TS requirement is not critical in knowing the status of the plant. Should some equipment be discovered inoperable, the extended AOT provides for some opportunity to restore the status to operable.

The deletion of a surveillance requirement that requires performing a vendor recommended maintenance at a specific frequency does not impact the ability of the EDG to perform its intended function for the mission time assumed in the accident analysis. EDG maintenance will continue to be performed and controlled under station procedures. The risk associated with the maintenance will be assessed under the provisions of 10 CFR 50.65 [Requirements for monitoring the effectiveness of maintenance at nuclear power plants], section (a) 4. The TS frequency was initially established to coincide with refueling outages, the only time that one EDG could be inoperable for any extended time. However, multiple plants have extended the time between refueling outages to 24 months with no discernable impact on reliability or availability. In addition, the Fairbanks-Morse diesel engine owners group has evaluated the maintenance requirements and determined that the TS required frequency should be based on performance and inspection results, not an arbitrary period that coincides with the best opportunity to perform the work. The Maintenance Rule requires evaluation for additional corrective actions and increased monitoring for scoped systems if the reliability and/or availability fall below pre-established criteria. This approach ensures appropriate actions in a timely manner are taken to ensure that equipment relied upon for accident mitigation is available when required.

There are no changes in operational limits or physical design of the onsite electric power systems. The proposed changes do not change the function or operation of plant equipment or affect the response of the equipment if called upon to operate. The EDGs are not the initiators of previously evaluated accidents. The EDGs are designed to mitigate the consequences of accidents. The risk informed assessment that was performed concluded that the increase in plant risk is small and consistent with the guidance in Regulatory Guide 1.174, [''An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the licensing Basis'']. This assessment considers the possibility of an accident occurring during the extended period that the EDG would be unavailable. The proposed changes allow for additional operational flexibility and will not cause a significant increase in the probability or consequences of an accident previously evaluated. In actuality, the installation and availability of the AAC will have an overall net reduction in core damage frequency.

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

No.

The proposed change to extend the EDG AOT to 14 days is based upon the installation of an AAC power source and the significant reduction in core damage frequency that results. There are no significant changes in installed plant equipment or operation of safety related equipment. The accident analysis considered the credible accidents and bounded those that apply.

The installation of the AAC and the extended AOT for one EDG to be inoperable remain bounded by previous evaluations.

The AOT extension to provide additional time to perform the redundant equipment verification is based on the other methods available for the Control Room staff to be made aware of a change in ESF equipment status and the safety benefit of performing this verification in an unhurried manner. This verification has been extended by other plants, both those who have converted to ITS and those that have not. No plant modifications are required and operator training is unaffected. The verification process does not utilize any new or complex software and any new accident is bounded by a Loss of Site Power or Station Blackout analysis.

The deletion of a surveillance requirement to perform the manufacturer's recommended inspection and maintenance is based on the recommendations from the vendor and the Fairbanks Morse owners group. The recommendation is to continue to perform the inspections and maintenance but the frequency should not be based on the refueling outage frequency. The effectiveness of the maintenance will be assured through monitoring under the Maintenance Rule program which would require evaluation and corrective actions should the EDG not meet its performance criteria for reliability and availability.

The EDG performs a function of supplying power when the normal ESF sources are unavailable. This is a function that mitigates the effects of the event and the proposed changes cannot cause the possibility of an accident that was not previously evaluated.

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

No.

The proposed change to extend the EDG AOT to 14 days from the current 72 hours will assure that an alternative source of power for the ESF onsite distribution system is available and ready. The AAC and interfacing equipment are designed to maintain independence and separation, particularly during faulted conditions. The plant equipment will continue to respond per the design and analysis. The performance capability of the EDGs will not be affected. Installation of the AAC will have a net reduction in the core damage frequency. In addition, administrative controls will ensure that there are adequate compensatory measures that can and will be taken during extended EDG maintenance activities to reduce overall risk.

The AOT extension to provide additional time to perform the redundant equipment verification for operability verification allows some time to discover a problem and make a minor repair prior to placing the plant in a shutdown transient. The types of corrective or preventative maintenance associated with an EDG will not change. Plant operating and emergency procedures will be enhanced with guidance on when to use the AAC and how to connect up to the ESF bus.

The deletion of the periodic EDG inspection per the vendor's recommendation at a proscribed frequency provides significant flexibility in when to schedule the inspection and preventative maintenance. The activities would still be performed but the frequency would be based on equipment performance and owners group recommendation. The plant analysis only considers the availability of the EDG. The TS surveillances that assure the EDG remains operable remain in place at their current frequencies and the maintenance requirement will assure that the EDG receives sufficient maintenance to remain operable.

Since the operation of the plant remains largely unaffected and the EDG or the AAC will supply power to the ESF equipment as needed, there is no significant reduction in a margin of safety.

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

Attorney for licensee: J. Hamilton Hagood, Jr., South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Evangelos C. Marinos.

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 29, 2005.

Description of amendment request: The proposed amendment would add requirements to TS 3/4.7.1.2 to assure continued operability of the Emergency Feedwater (EFW) System based on LER 1998–004–00, by including the newly installed six emergency feedwater system automatic isolation valves into the Surveillance Requirements to assure the capability for automatic isolation of EFW in the event of a faulted steam generator.

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

1. Does the proposed change involve a significant increase in the probability or

consequences of an accident previously evaluated?

No.

The proposed change addresses necessary changes to the VCSNS [Virgil C. Summer Nuclear Station] Technical Specification (TS) 4.7.1.2.b and 4.7.1.2.c.2 associated with the installation of six new automatic isolation valves in the EF[W] system.

The only Final Safety Analysis Report (FSAR) analyzed accident for which the EF[W] system could contribute as an initiator would be minor secondary line break, as described in Section 15.3.2. The addition of isolation valves in the EF[W] piping to the steam generators [SGs] will not increase the likelihood of a pipe break, since the addition will be in accordance with the same codes and standards as the corresponding, existing portions of the system. Piping stress analyses have demonstrated the addition of these valves does not result in the need to postulate any additional pipe breaks.

The accidents analyzed in the FSAR, which rely on EF[W system] to mitigate consequences, are loss of normal feedwater, loss of off-site power, and major secondary system pipe ruptures. The addition of these automatic isolation valves will eliminate the need for operator action to manually close a flow control valve in response to a major secondary system line break. The elimination of operator manual action is accomplished by the addition of a new pneumatically operated isolation valve in series with each of the six existing flow control valves.

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 previously evaluated?

No.

This proposed change does not result in changes to actual operating pressures, flow rates, flow paths, or system interfaces. There are no alterations to system operability requirements. The existing system alarm set points are not affected, neither is the information available to the operators. The addition of six new isolation valves will not change system design criteria and the surveillance testing will be the same as for the existing flow control valves.

This change does not introduce any new or different kind of failure mechanisms or limiting single failures. Piping analysis has concluded that no new pipe break locations or break sizes will result from this change. Equipment protection features are not impacted, the frequency of pump and valve operation remains the same. Independence and redundancy are actually improved.

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

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

No.

The design basis for the EF[W] system is to assure the required flow and pressure to remove decay heat from the core under the worst postulated conditions. An additional function of the system is to isolate flow to a faulted SG within the time assumed in the safety analysis. The proposed change eliminates the need for operators to take actions to manually close the flow control valves in the event of a single failure.

The proposed change will create a surveillance requirement for the new isolation valves that is the same as the existing flow control valves. The acceptance criteria will assure the operability of these valves. The design and installation of these isolation valves will maintain the requirements for independence, redundancy, separation and testability. The margins assumed in the safety analysis will be enhanced by this proposed change. Due to the automatic isolation capability, additional water will be available for the intact SGs and a reduced mass will be available to be released into the containment building. No credible single failure will be capable of preventing isolation of a faulted SG upon a high flow signal.

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

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

Attorney for licensee: J. Hagood Hamilton, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Branch Chief: Evangelos C. Marinos.

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 29, 2005.

Description of amendment request: This amendment revises Technical Specification (TS) 6.9.1.5 and TS 6.9.1.10 by eliminating the requirements to submit monthly operating reports and occupational radiation exposure reports. This consolidated line item improvement process (CLIIP) TS change was noticed in the **Federal Register** on June 23, 2004, (69 FR 35067). In addition, the TSs are revised beyond the scope of the CLIIP by the deletion of the TS 6.9.15 requirement to report exceedence of coolant specific activity limits and an administrative change to a TS index page.

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:

SCE&G has reviewed the proposed no significant hazards consideration determination published on June 23, 2004 (69 FR 35067) as part of the CLIIP. SCE&G has concluded that the proposed determination presented in the notice is applicable to the VCSNS, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

The deletion of the additional paragraph in 6.9.1.5 is beyond the scope of the CLIIP and as such is beyond the scope of the no significant hazards consideration determination published on June 23, 2004. Therefore the following evaluation has been performed.

In accordance with the criteria set forth in 10 CFR 50.92, SCE&G has evaluated the proposed beyond scope Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided to support this conclusion.

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

No.

The proposed change is the deletion of a paragraph in the administrative controls section of the facility Technical Specifications. The paragraph identifies required information that was to be provided in a report to the staff in the event where the RCS specific activity exceeded TS limits. This report has been found to be unnecessary due to reporting requirements located in 10 CFR 50.73 (exceeding a TS limit). Additionally, the TS limits are set such that there is very little risk to the health and safety of the public. Before the condition became significant, the NRC would have been notified due to the 10 CFR 50.73 requirement to report significant degradations in a principal fission product barrier.

Deletion of an administrative controls paragraph that provides reporting requirements is not a precursor to an accident. No changes are being proposed to any installed plant equipment or procedures. The operating philosophy is unaffected and training is not impacted. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No.

The proposed change is the deletion of a paragraph that was inserted per the guidance of Generic Letter 85–19. The staff was concerned that the reporting requirements prior to that time were too restrictive and relaxed them through the Generic Letter. Since that time, it was determined that specific reporting could be performed via requirements in 10 CFR 50.73. Exceeding the TS limit is now an uncommon condition as proper fuel management and fabrication techniques should preclude approaching the TS limit.

Revising or even deleting a reporting requirement in the facility TS will not impact how the plant is operated, how data is evaluated, or what instructions are located in operating and emergency procedures. No new equipment is being installed and no plant modifications are resulting from this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No.

The proposed change to delete some specific reporting requirements in the Administrative Controls section of TS has no impact on any plant evaluation or analysis. No plant setpoints are impacted; no alarm or annunciator functions are affected. This change has been approved for other plants. 10 CFR 50.73 will still require reporting the condition should it ever occur. 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: J. Hagood Hamilton, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Evangelos C. Marinos.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: September 19, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Limiting Conditions for Operation (LCO) 3.3.1, ''Reactor Trip system (RTS) Instrumentation'' and TS Surveillance Requirements (SR) 3.2.4.2, ''Quadrant Power Tilt Ration (QPTR)'' to avoid confusion as to when a flux map for QPTR is required.

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

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

No.

The proposed changes 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 within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences. Therefore, the proposed changes do not increase 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 previously evaluated?

No.

The proposed changes do not result in a change in the manner in which the RTS and ESFAS provide plant protection. The RTS and ESFAS will continue to have the same set points after the proposed changes are implemented. There are no design changes associated with the license amendment.

The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed 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?

No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. 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.

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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Branch Chief: Evangelos C. Marinos.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 16, 2005.

Brief description of amendments: The proposed change would revise Technical Specifications (TSs) 3.3.2, ''ESFAS [Engineered Safety Features Actuation System] Instrumentation''; 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating''; and 3.6.7, ''Spray Additive System.''

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

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

Response: No.

None of the changes impact the initiation or probability of occurrence of any accident.

The consequences of accidents evaluated in the FSAR [Final Safety Analysis Report] that could be affected by this proposed change are those involving the pressurization of the containment and associated flooding of the containment and recirculation of this fluid within the ECCS or the Containment Spray System (*e.g.*, LOCAs [loss-of-coolant accidents]).

Although the water level in the containment flood plain will be higher at the start of ECCS switchover, the maximum water levels observed for the duration of the accident are unchanged by the nominal setpoint changes.

The increase in the minimum water delivered to containment by the RWST [Refueling Water Storage Tank] setpoint change will reduce the radiological consequences of LOCAs by diluting the radioiodine concentrations in the recirculating sump fluid which could be released by Engineered Safety Features (ESF) leakage. This increase in water will also reduce the maximum pH and its deleterious effects on equipment and sump performance.

The increase in water level and the change in strainer design will significantly increase NPSH [net positive suction head] and headloss margins required to assure long term core cooling.

The change to a minimum pH of 7.1 will not result in a significant increase in the radiological consequences of a LOCA as described below.

The buffering agent will dissolve in the containment sump fluid resulting from these accidents raising the pH of the fluid, which would initially be greater than or equal to 4.0 but less than 7.0 during the injection phase of containment spray operation. The equilibrium spray pH during the recirculation phase resulting from this change will be greater than or equal to 7.1. The pH range for the spray will be bounded by the water spray solution which is borated water with a maximum of 2600 ppm [parts per million] boron buffered to a final spray solution pH much less than the 10.5 as described in the current FSAR Section 3.11(B) for the postulated spray solution environment. The maximum pH is the limiting parameter for equipment qualification. Since the resulting pH level will be closer to neutral using the lower limit of 7.1, post-LOCA corrosion of containment components will not be increased. Post-LOCA hydrogen generation will be reduced. There will not be an adverse radiation dose effect on any safety-related equipment. Thus, the potential for failures of the ECCS or safety-related equipment following a LOCA will not be increased as a result of the proposed change.

This modification affects the Containment Spray System which is intended to respond to and mitigate the effects of a LOCA. The chemical additive baskets serve a passive function to provide a buffering agent to neutralize the sump solution. Failure of a basket would not initiate an accident. The Containment Spray System will continue to function in a manner consistent with the plant design basis. There will be no degradation in the performance of nor an increase in the number of challenges to equipment assumed to function during an accident situation.

As such, these Technical Specification revisions do not affect the probability of any event initiators. There will be no adverse changes to normal plant operating parameters, ESF actuation setpoints, or accident mitigation capabilities.

The proposed change allows a passive Spray Additive System to replace the active Spray Additive System currently used to mitigate the consequences of an accident. By substituting a passive system for an active system, the probability of occurrence of a malfunction of equipment associated with the Spray Additive System will be reduced since the number of active components subject to malfunction is reduced. This TS surveillance change will maintain the equilibrium sump pH at greater than or equal to 7.1 to minimize chloride-induced stress corrosion cracking in austenitic stainless components important to safety located inside containment. Therefore, the proposed changes will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

The offsite and control room doses will continue to meet the requirements of 10 CFR [Part] 100; 10 CFR [Part] 50, Appendix A, GDC [General Design Criterion] 19; SRP [Standard Review Plan] 15.6.5.11; and SRP 6.4.11. The deletion of the active Spray Additive System and replacement with a sump pH control system using TSP–C [Trisodium Phosphate crystalline] will not increase the reported radiological consequences of a postulated LOCA. The proposed new pH control system will provide satisfactory retention of iodine in the sump water, as well as provide adequate pH control to minimize the potential of chloride-induced stress corrosion cracking of austenitic stainless steel components.

The baskets which will contain the trisodium phosphate are seismically designed and located in the post-accident flood plane area to ensure mixing with the recirculating fluid. The consequences of a malfunction of any piece of equipment associated with the Containment Spray System would not be affected by the change from an active Spray Additive System to a passive system. The consequences of a failure in the active Spray Additive System are eliminated by this passive system. The proposed changes do not increase the malfunction of equipment important to safety previously evaluated in the FSAR. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The changes to the new Containment Spray Additive System are essentially a passive system, i.e., no operator or automatic action of electrical devices is required to actuate the system. There are no electrical components being added whose failure could prevent the new system from functioning. The only new components being added are the storage baskets for the chemical buffering agent. Seismic requirements have been included in the design to ensure the structural integrity of the baskets will be maintained during a seismic event.

No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. The use of dry sodium phosphates is allowed for adjustment of the post-LOCA sump solution pH as discussed in SRP 6.1.1. The quantity of trisodium phosphate or any other buffering agent chosen will provide a minimum equilibrium sump pH of 7.1 following dissolution and mixing. Therefore, the possibility of a new or different type of accident is not created.

There are no changes which would cause the malfunction of safety-related equipment, assumed to be operable in the accident analyses, as a result of the proposed Technical Specification changes. No new equipment performance burdens are imposed; however, there is the potential for an unlikely, but possible, event in which an initially concentrated solution of buffering agent could be transported to the stagnant volume of an inactive sump during blowdown and pool fill. This situation would be short-lived since, as the recirculated sump fluid is cooled in the RHR [residual heat removal] heat exchangers, sufficient buoyancy-driven circulation within containment will result to displace the stagnant solution and eventually yield a uniform, equilibrium solution. In the current design, all of the chemical additive is delivered to the recirculation sump even in the event of the worst single active failure. The possibility of a malfunction of safety-related equipment with a different result is not created. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The RWST Low-Low nominal setpoint, in conjunction with the plant modifications, ensures that both the ECCS and Containment Spray Systems can be transferred from injection to recirculation without stopping the pumps and with no credit for containment overpressure. Analyses have been performed which show that, even with worst case single active failures, suction to the pumps would not be lost.

The only function of the NaOH spray additive solution is to provide pH control of the post-accident containment recirculation sump water, since the borated water from the Refueling Water Storage Tank (RWST) used as the containment spray pump suction source during injection is sufficient to remove iodine from the containment atmosphere following a LOCA. The net effect on the pH control function of reducing the amount of NaOH or replacing NaOH with the chemical buffering agent TSP–C is that the equilibrium sump pH will be lowered to a minimum of 7.1. There will be no change to the current Technical Specification acceptance limits on RWST volume and boron concentration. The resulting equilibrium sump pH level from this change will be closer to neutral; therefore, the post-LOCA corrosion of containment components will not be increased.

Because the long term pH will be maintained greater than or equal to 7.1, margin to minimize the potential for stress corrosion cracking is maintained.

The radiological analysis as discussed in the technical analysis above, is shown not to be impacted. There will be no change to the DNBR [departure from nucleate boiling ratio] Correlation Limit, the design DNBR limits, or the safety analysis DNBR limits discussed in Bases Section 2.1.1. 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. There will be no adverse impact on DNBR limits, F_Q, F-delta-H, LOCA PCT [peak cladding temperature], peak local power density, or any other margin of safety. Therefore the proposed change does not involve a 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: David Terao.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 16, 2005.

Brief description of amendments: The amendment would revise the Technical Specifications (TS) to adopt NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The proposed amendment includes:

—Revised TS definition of Leakage,
—Revised TS 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,''
—Added new TS 3.4.17, ''Steam Generator (SG) Tube Integrity,''
—Revised TS 5.5.9, ''Steam Generator Program''
—Added new TS 5.6.9, ''Steam Generator Tube Inspection Report,'' and
—Revised TS 5.6.10, ''Steam Generator Tube Inspection Report'' (for existing Unit 1 SGs).

The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) Report 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 December 16, 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 steam line break], rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

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

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The 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 150 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 type of accident from any accident previously evaluated.

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

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

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

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: David Terao.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: December 16, 2005.

Brief description of amendments: The proposed amendments would revise the Technical Specifications (TSs) consistent with the Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveller, TSTF–419, ''Revise PTLR [Pressure and Temperature Limits Report] Definition and References in ISTS [improved Standard TS] 5.6.6.

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

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

Response: No.

The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T [Pressure/Temperature] limits or LTOP [Low Temperature Overpressure Protection] setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The proposed changes 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 within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. 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. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

Response: No.

The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: David Terao.

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

Date of amendment request: January 31, 2006.

Description of amendment request: The proposed change would replace the current containment methodology with the methodology described in Topical Report DOM–NAF–3, ''GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment,'' increase the containment air partial pressure limits in Technical Specification (TS) 3.8, ''Containment,'' revise the loss-of-coolant (LOCA) accident alternate source term (AST) analysis, and change the method of starting the recirculation spray (RS) pumps.

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

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

No.

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

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

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

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

No.

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

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

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

No.

The changes to the actuation of the RS pumps and the increased containment air partial pressure affect the containment response analyses and the LOCA dose analysis. Analyses have been performed that show the containment design basis limits are satisfied and the post-LOCA offsite and control room doses meet the required criteria for the proposed changes to the containment analysis methodology, the RS pump start method, the TS containment air partial pressure limits, and the LOCA AST bases. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Evangelos C. Marinos.

Notice of Issuance of Amendments to Facility Operating Licenses

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

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

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

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

Date of application for amendment: February 25, 2005.

Brief description of amendment: The amendment deleted Section 2.E of the Facility Operating License, which requires reporting of violations of the requirements in Section 2.C of the Facility Operating License.

Date of Issuance: February 22, 2006.

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

Amendment No.: 258.

Facility Operating License No. DPR–16: The amendment revised the Facility Operating License.

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21453).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 22, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 4, 2005, as supplemented by letter dated January 25, 2006.

Brief description of amendments: The proposed amendments deleted Section 2.F (2.G in Unit 3) of the Facility Operating Licenses, which requires reporting violations of the requirements in Section 2.C of the Facility Operating License. The amendments also make administrative and editorial changes to the Technical Specifications (TSs). Changes to TS 1.4, ''Frequency,'' and TS 3.4.3, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' correct editorial errors. The changes to TS 2.1.1, ''Reactor Core SLs [Safety Limits],'' and TS 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operating,'' remove the reference to departure from nucleate boiling ratios (DNBR) based on operating cycle, since only one of the listed DNBR values is now valid. TS 3.1.10, ''Special Test Exceptions (STE)—MODES 1 and 2,'' is changed to correct an inconsistency between the limiting condition for operation and the TS Bases. The changes to TS 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' and TS 3.7.3, ''Main Feedwater Isolation Valves (MFIVs),'' correct the applicability for these specifications. The change to TS 3.8.1, ''AC [Alternating Current] Sources—Operating,'' adds a note to a surveillance requirement. Changes to TS 3.8.4, ''DC [Direct Current] Sources—Operating,'' and TS 3.8.6, ''Battery Cell Parameters,'' remove the reference to AT&T batteries. The changes to TS 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' correct the reference for NRC notification.

Date of issuance: February 28, 2006.

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

Amendment Nos.: Unit 1—158, Unit 2—158, Unit 3—158.

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

*Date of initial notice in **Federal Register:*** May 10, 2005 (70 FR 24647).

The January 25, 2006, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 31, 2005.

Brief description of amendment: The amendment modified requirements by adding to the technical specifications a Limiting Condition for Operation (LCO) 3.0.8 that provides 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. In addition, a change to LCO 3.0.1 was required to reference the addition of LCO 3.0.8.

Date of issuance: February 15, 2006.

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

Amendment No.: 172.

Facility Operating License No. NPF–43: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72670).

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

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 letter dated January 17, 2006.

Brief description of amendment: The amendment allows a one-time extension of an additional 4 months beyond the 5-year extension already granted by the staff to the nominal 10-year interval of the test interval for the next Appendix J, Type A test.

Date of issuance: February 9, 2006.

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

Amendment No.: 150.

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

*Date of initial notice in **Federal Register:*** March 29, 2005 (70 FR 15942). The supplement dated January 17, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 30, 2005, as supplemented by letter dated November 21, 2005.

Brief description of amendment: The amendment incorporated the following U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) changes that apply to the Boiling Water Reactor/6 Improved Standard Technical Specifications into GGNS Technical Specifications (TSs):

TSTF No.	Description	TS section affected
TSTF–046, Rev. 1	Clarify the Containment Isolation Valve surveillance to apply only to automatic isolation valves.	Surveillance Requirement (SR) 3.6.1.3.4, SR 3.6.4.2.2, SR 3.6.5.3.3.
TSTF–222, Rev. 1	Control Rod Scram Time Testing	SR 3.1.4.1, SR 3.1.4.4.
TSTF–264, Rev. 0	Delete flux monitors specific overlap SRs	SR 3.3.1.1.5, SR 3.3.1.1.6, Table 3.3.1.1–1.
TSTF–275, Rev. 0	Clarify requirements for Diesel Generator (DG) start signal on Reactor Pressure Vessel (RPV) Level—Low, Low, Low during RPV cavity flood-up.	Table 3.3.5.1–1, Footnote (a).
TSTF–276, Rev. 2	Revise DG full load rejection test	SR 3.8.1.9, SR 3.8.1.10, SR 3.8.1.14.
TSTF–300, Rev. 0	Eliminate DG Loss of Coolant Accident (LOCA) Start SRs while in shutdown when Emergency Core Cooling System is not required.	SR 3.8.2.1.
TSTF–322, Rev. 2	Secondary Containment Integrity SRs	SR 3.6.4.1.3, SR 3.6.4.1.4.
TSTF–400, Rev. 1	Clarify SR on bypass of DG automatic trips	SR 3.8.1.13.

TSTF No.	Description	TS section affected
TSTF–416, Rev. 0	SR 3.5.1.2 Notation	Limiting Condition for Operation (LCO) 3.5.1, SR 3.5.1.2, LCO 3.5.2, SR 3.5.2.4.

The amendment also granted delayed performance of the modified SRs for DG 12 until the next regularly scheduled performance rather than immediately upon implementation of this amendment, which is still consistent with NRC-approved TSTF changes. Those SRs are SR 3.8.1.9, SR 3.8.1.10, and SR 3.8.1.14.

Date of issuance: February 2, 2006.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance, with the exception of SR 3.8.1.9, SR 3.8.1.10, and SR 3.8.1.14.

Amendment No: 169.

Facility Operating License No. NPF–29: The amendment revises the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29791). The supplemental letter dated November 21, 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 February 2, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

Date of application for amendment: February 25, 2005.

Brief description of amendment: The amendment deleted the reporting requirement in the Renewed Facility Operating License related to reporting violations of other requirements in the operating license.

Date of issuance: February 17, 2006.

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

Amendment Nos.: 210.

Facility Operating License No. DPR–19: The amendments revised the Facility Operating License.

Date of initial notice in ***Federal Register:*** April 26, 2005 (70 FR 21456).

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: April 13, 2005, as supplemented by letter dated December 22, 2005.

Brief description of amendments: The amendment extended the completion time (CT) for Required Action A.1, ''Restore Residual Heat Removal Service Water subsystem to OPERABLE status,'' associated with Technical Specification (TS) Section 3.7.1 from 7 days to 10 days; established a 6-day (for Division 2 core standby cooling system (CSCS) maintenance) or 10-day (for Division 1 CSCS maintenance) CT for TS Section 3.7.2 when one or more required diesel generator cooling water subsystem(s) are inoperable. The Nuclear Regulatory Commission (NRC) staff is granting this amendment request with respect to TS Sections 3.7.1 and 3.7.2 only. In the original submittal, the licensee also requested an extension of the CT for required Action C.4, ''Restore required Diesel Generator (DG) to OPERABLE status,'' associated with TS 3.8.1 from 72 hours to 6 days; and extension of the CT for required Action F.1, ''Restore one required Diesel Generator (DG) to OPERABLE status,'' associated with TS 3.8.1 from 2 hours to 6 days. The NRC staff needs additional information from the licensee in order to complete its review and grant this portion of the amendment request. The staff will address the requests to extend CTs for TS 3.8.1 in a separate safety evaluation and license amendment, if granted.

Date of issuance: February 23, 2006.

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

Amendment Nos.: 175/161

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 7, 2005 (70 FR 33213).

The December 22, 2005, supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: December 14, 2005, as supplemented by letter dated February 13, 2006.

Brief description of amendment: The amendment modifies the Technical Specifications (TSs) to incorporate a revised Single Loop Operation Safety Limit Minimum Critical Power Ratio due to the cycle-specific analysis.

Date of issuance: March 1, 2006.

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

Amendment No.: 183.

Facility Operating License No. NPF–39 This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** January 17, 2006 (71 FR 2590). The supplement dated February 13, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 17, 2004.

Brief description of amendments: The amendments revised Appendix B, Environmental Protection Plan (non-radiological), of the Limerick Generating Station Operating Licenses.

Date of issuance: February 17, 2006.

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

Amendment Nos.: 180 and 142.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Environmental Protection Plan.

Date of initial notice in ***Federal Register:*** April 12, 2005 (70 FR 19112).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 25, 2005.

Brief description of amendments: The proposed amendment would delete the sections of the Facility Operating Licenses that require reporting of violations of the requirements in Section 2.C of the Facility Operating Licenses.

Date of issuance: February 17, 2006.

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

Amendment Nos.: 181 and 143.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications/license.

Date of initial notice in ***Federal Register:*** April 26, 2005 (70 FR 21457).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 21, 2005.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) by relocating the Pressure Isolation Valve Table to the Technical Requirements Manual.

Date of issuance: February 17, 2006.

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

Amendment Nos.: 182 and 144.

Facility Operating License Nos. NPF–39 and NPF–85. These amendments revised the TSs.

Date of initial notice in ***Federal Register:*** January 17, 2006 (71 FR 2590).

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

No significant hazards consideration comments received: No.

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 application for amendments: June 2, 2004, as supplemented February 11, May 12, October 31, and November 14, 2005.

Brief description of amendments: These amendments approve conversion of the BVPS–1 and 2 containments from subatmospheric to atmospheric operating conditions. The proposed changes also approves the Modular Accident Analysis Program—Design Basis Accident (MAAP–DBA) computer code for the BVPS–1 and 2 containment integrity analysis and changes to mass and energy calculation methodologies.

Date of issuance: February 6, 2006.

Effective date: For BVPS–1, the amendment is effective as of the date of its issuance and shall be implemented prior to Mode 4 entry during startup from 1R17 which begins on or about February 10, 2006. For BVPS–2, the amendment is effective as of the date of its issuance and shall be implemented prior to Mode 4 entry during startup from 2R12 which begins October 2006.

Amendment Nos.: 272 and 154.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 20, 2004 (69 FR 43462).

The supplements dated February 11, May 12, October 31, and November 14, 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 February 6, 2006.

No significant hazards consideration comments received: No.

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 application for amendments: February 11, 2005, as supplemented August 8, 2005.

Brief description of amendments: The amendments approved the adoption of the Relaxed axial offset control (RAOC) and F_Q surveillance methodologies in accordance with NRC-approved Topical Report WCAP–10216–P–A, ''Relaxation of Constant Axial Offset Control—F_Q Surveillance Technical Specification.'' TS 3.2.1, ''Axial Flux Difference (AFD),'' and TS 3.2.2, ''Heat Flux Hot Channel Factor—$F_Q(Z)$,'' were revised to adopt the RAOC calculational procedure of NUREG–1431, ''Standard Westinghouse Technical Specifications for Westinghouse Plants,'' Revision 3, June 2004. Changes to TS 3.2.3, ''Nuclear Enthalpy Hot Channel Factor—$F^N\Delta_H$,'' TS 3.2.4, ''Quadrant Power Tilt Ratio (QPTR),'' TS 3.3.1, ''Reactor Trip System Instrumentation (Table 4.3–1, Note 3),'' and TS 6.9.5, ''Core Operating Limits Report (COLR),'' were made to provide consistency with the changes made to TSs 3.2.1 and 3.2.2.

Date of issuance: February 27, 2006.

Effective date: Prior to entry into Mode 4 upon restart from the spring 2006 refueling outage which begins on or about February 10, 2006, for BVPS–1 and prior to entry into Mode 4 from startup following the fall 2006 refueling outage which begins in October 2006, for BVPS–2.

Amendment Nos.: 274 and 155.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 26, 2005 (70 FR 21457). The supplement dated August 8, 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 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 February 27, 2006.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: April 20, 2005.

Brief description of amendment: The changes revised the Technical Specifications (TSs) to replace plant-specific position titles with generic position titles. Also, the changes deleted TS 6.7, ''Safety Limit Violations or Protective Limit Violation,'' and included a change to TS 2.1.2, ''Reactor Core,'' associated with the deletion of TS 6.7. Additionally, the changes relocated to the Davis-Besse Nuclear Power Station Updated Safety Analysis Report the Process Control Program requirements from TS 6.8, ''Procedures and Programs,'' and from TS 6.14, ''Process Control Program (PCP).'' Associated with this change, TS Definition 1.30, ''Process Control Program,'' was deleted. Also, TS 6.15, ''Offsite Dose Calculation Manual (ODCM),'' was modified to eliminate the requirement that changes to the ODCM be reviewed and accepted by the Plant Operations Review Committee (PORC). These changes to administrative requirements also eliminated the need to propose additional changes in the future to plant-specific position/organizational titles. The changes are consistent with NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 3, dated June 2004. Lastly, the changes revised in the TSs the title ''Industrial Security Plan'' to ''Physical Security Plan.''

Date of issuance: February 7, 2006.

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

Amendment No.: 272.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 24, 2005 (70 FR 29795).

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

No significant hazards consideration comments received: No.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 28, 2005.

Description of amendment request: The amendment revised the Seabrook Station, Unit No. 1, Technical Specifications (TSs) Surveillance Requirement 4.1.1.3, ''Moderator Temperature Coefficient,'' to allow the option of not measuring the moderator temperature coefficient within 7 effective full-power days of reaching an equilibrium boron concentration of 300 parts per million. This option is available only if the conditions described in WCAP–13749–P–A, ''Safety Evaluation Supporting the Conditional Exemption of the Most Negative Moderator Temperature Coefficient Measurement'' have been met.

Date of issuance: February 17, 2006.

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

Amendment No.: 107.

Facility Operating License No. NPF–86: The amendment revised the TSs.

*Date of initial notice in **Federal Register:*** May 10, 2005 (70 FR 24652).

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

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–388, Susquehanna Steam Electric Station, Unit 2 (SSES–2), Luzerne County, Pennsylvania

Date of application for amendment: January 28, 2005.

Brief description of amendment: The amendment revises the SSES–2 Technical Specification (TS) Table 3.3.5.1–1, ''Emergency Core Cooling System Instrumentation,'' Function 3.e, '' High Pressure Coolant Injection (HPCI) System,'' to change Condition ''D'' to ''C'' as the condition to reference from Required Action A.1. This is an editorial revision to correct a typographical error that had been present since the conversion to the Improved TSs in July 1998.

Date of issuance: February 6, 2006.

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

Amendment No.: 206.

Facility Operating License No. NPF–22: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 10, 2005 (70 FR 24654).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 2, 2006.

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 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of application for amendments: February 7, 2005

Brief description of amendments: The amendments change the SSES 1 and 2 Technical Specifications (TSs) for ''Secondary Containment,'' limiting condition for operation 3.6.4.1, by revising the frequency note applicable to Surveillance Requirements (SR) 3.6.4.1.4 and SR 3.6.41.5. The revised note requires each zone configuration be tested at least once every 60 months.

Date of issuance: February 2, 2006.

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

Amendment Nos.: 229 and 205.

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 February 2, 2006.

No significant hazards consideration comments received: No.

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 application for amendments: April 26, 2004, as supplemented by letters dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005.

Brief description of amendments: These amendments revised the Technical Specifications to incorporate a full-scope application of an alternate source term methodology in accordance with 10 CFR 50.67.

Date of issuance: February 17, 2006.

Effective date: As of the date of issuance, to be implemented with 90 days.

Amendment Nos.: 271 and 252.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 22, 2004 (69 FR 34705). The supplements did not effect the scope of changes discussed in the original no significant hazards determination.

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

No significant hazards consideration comments received: No.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: April 29, 2005, as supplemented on September 19, 2005.

Brief description of amendment: The amendment revised the Technical Specifications to incorporate the relaxed axial offset control and heat flux hot channel (FQ) surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control requirements and to increase the ability to return to power after a plant trip or transient. The changes are consistent with Westinghouse Electric Company Report WCAP–10216–P–A, ''Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification.''

Date of issuance: February 15, 2006.

Effective date: As of the date of issuance to be implemented prior to startup following the fall 2006 refueling outage.

Amendment No.: 94.

Renewed Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 7, 2005 (70 FR 33220).

The September 19, 2005, letter 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 February 15, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 15, 2005, and as supplemented by letter dated January 20, 2006.

Brief description of amendments: The amendments are for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, operating licenses, but they involved Unit 1, which is not an operating nuclear plant and is in the process of being decommissioned. The amendments revised License Condition 2.B.(6) for both SONGS, Units 2 and 3, by (1) deleting the sentence ''Transshipment of Unit 1 fuel between Units 1 and [2 or 3] shall be in accordance with SCE [Southern California Edison Company] letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71'' and (2) adding the phrase ''and by the decommissioning of San Onofre Nuclear Generating Station Unit 1'' to the remaining sentence in the license condition. This change recognized that Unit 1 is now in the stage of decommissioning and that in the future any radioactive waste water produced in the further decommissioning of Unit 1 would be released from the San Onofre site by transferring the waste water from Unit 1 to Units 2 and 3. The processing (if required) and discharging of this waste water would be using the Units 2 and 3 radioactive waste system and ocean outfall discharge line.

Date of issuance: February 28, 2006.

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

Amendment Nos.: Unit 2—202; Unit 3—193.

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

*Date of initial notice in **Federal Register:*** September 13, 2005 (70 FR 54089).

The supplement dated January 20, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2006.

No significant hazards consideration comments received: No.

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 amendments request: November 2, 2005.

Brief Description of amendments: The amendments modify technical specifications (TS) to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, ''Increased Flexibility in Mode Restraints.''

Date of issuance: February 22, 2006.

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

Amendment Nos.: 170 and 163.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 20, 2005 (70 FR 75498).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 7th day of March, 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. 06–2383 Filed 3–13–06; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Public Availability of Fiscal Year 2005 Agency Inventories Under the Federal Activities Inventory Reform Act

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Notice of public availability of agency inventory of activities that are not inherently governmental and of activities that are inherently governmental.

SUMMARY: The Federal Activities Inventory Reform (FAIR) Act, Public Law 105–270, requires agencies to develop inventories each year of activities performed by their employees that are not inherently governmental—i.e., inventories of commercial activities. The FAIR Act further requires OMB to review the inventories in consultation with the agencies and publish a notice of public availability in the **Federal Register** after the consultation process is completed. In accordance with the FAIR Act, OMB is publishing this notice to announce the availability of inventories from the agencies listed below. These inventories identify both commercial activities and activities that are inherently governmental.

This is the first release of the FAIR Act inventories for FY 2005. Interested parties who disagree with the agency's initial judgment may challenge the inclusion or the omission of an activity on the list of activities that are not inherently governmental within 30 working days and, if not satisfied with this review, may appeal to a higher level within the agency.

The Office of Federal Procurement Policy has made available a FAIR Act User's Guide through its Internet site: *http://www.whitehouse.gov/omb/procurement/fair-index.html*. This User's Guide will help interested parties review FY 2005 FAIR Act inventories.

Joshua B. Bolten,

Director.

FIRST FAIR ACT RELEASE FY 2005

American Battle Monuments Commission	Mr. Alan Gregory, (703) 696–6868, www.abmc.gov.
Chemical Safety Board	Ms. Bea Robinson, (202) 261–7627, www.csb.gov.