Environment Assessment (Closed—Ex. 1)

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Dated: November 3, 2005.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 05–22316 Filed 11–4–05; 11:02 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 14, 2005 to October 27, 2005. The last biweekly notice was published on October 25, 2005 (70 FR 61655).

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 01F21, 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.*

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Wake and Chatham Counties, North Carolina

Date of amendment request: August 18, 2005.

Description of amendment request: The amendment will allow the use of fire-resistive electrical cable, which has been demonstrated to provide an equivalent level of protection as would be provided by 3-hour and 1-hour rated electrical cable raceway fire barriers, for the protection of safe shutdown electrical cable.

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.

Operation of HNP in accordance with the proposed amendment does not increase the probability or consequences of accidents previously evaluated. The Final Safety Analysis Report (FSAR) documents the analyses of design basis accidents (DBA) at HNP. Any scenario or previously analyzed accidents that result in offsite dose were evaluated as part of this analysis. The proposed amendment does not adversely affect accident initiators nor alter design assumptions, conditions, or configurations of the facility. The proposed amendment does not adversely affect the ability of structures, systems, or components (SSCs) to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to assure that redundant trains of safe shutdown (SSD) control circuits remain protected from damage in the event of a postulated fire. The proposed amendment revises the Final Safety Analysis Report (FSAR) to use three-hour fire-resistive electrical cable, which has been demonstrated to provide an equivalent level of protection as would be provided by three-hour and one-hour rated electrical cable raceway fire barriers, for the protection of

SSD electrical cables. Based on the above, SSD control circuit protection is maintained by this amendment.

Therefore, this amendment 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.

Operation of HNP in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The FSAR documents the analyses of design basis accidents (DBA) at HNP. Any scenario or previously analyzed accidents that result in offsite dose were evaluated as part of this analysis. The proposed amendment does not change or affect any accident previously evaluated in the FSAR, and no new or different scenarios are created by the proposed amendment. The proposed amendment does not adversely affect accident initiators nor alter design assumptions, conditions, or configurations of the facility. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to assure that redundant trains of Safe Shutdown (SSD) control circuits remain protected from damage in the event of a postulated fire. The proposed amendment revises the Final Safety Analysis Report (FSAR) to use three-hour fire-resistive electrical cable, which has been demonstrated to provide an equivalent level of protection as would be provided by three-hour and one-hour rated electrical cable raceway fire barriers, for the protection of SSD electrical cables. Based on the above, SSD control circuit protection is maintained by this amendment.

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

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

Response: No.

Operation of HNP in accordance with the proposed amendment does not involve a significant reduction in a margin of safety. The proposed amendment does 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 affected by this change. The proposed amendment does not adversely affect existing plant safety margins or the reliability of equipment assumed to mitigate accidents in the FSAR. The proposed amendment does not adversely affect the ability of SSCs to perform their design function. SSCs required to safely shut down the reactor and to maintain it in a safe shutdown condition remain capable of performing their design functions.

The purpose of this amendment is to assure that redundant trains of Safe Shutdown (SSD) control circuits remain protected from damage in the event of a postulated fire. The proposed amendment revises the Final Safety Analysis Report (FSAR) to use three-hour fire-resistive electrical cable, which has been demonstrated to provide an equivalent level of protection as would be provided by three-hour and one-hour rated electrical cable raceway fire barriers, for the protection of SSD electrical cables. Based on the above, SSD control circuit protection is maintained by this amendment.

Therefore, this 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall, Jr.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: September 1, 2005.

Description of amendment request: The amendment will add Technical Specification (TS) 3.7.14, ''Fuel Storage Pool Boron Concentration'' and revise TS 5.6, ''Fuel Storage.'' The proposed changes are related to requirements for ensuring adequate subcriticality margin in the spent fuel storage pools. TS 5.6.1 is being revised to include the design requirements for dry storage of new fuel.

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 do not modify the facility. The accident previously analyzed for the spent fuel pool is a fuel handling accident. The proposed change applies administrative controls for maintaining the required boron concentration in the spent fuel storage pools, revises acceptance criteria and storage arrangements for fuel storage in PWR [pressurized-water reactor] ''flux trap'' style racks and adds acceptance criteria for dry storage of new fuel to the Technical Specifications. The controls on spent fuel pool boron and dry storage of new fuel have previously been implemented but are being added to the Technical Specifications as requirements. The proposed change applies new acceptance criteria for criticality safety of fuel storage in PWR ''flux trap'' style racks in Pools ''A'' and ''B.'' The new acceptance criteria require new administrative controls on the placement of fuel in Pools ''A'' and ''B.'' Similar administrative controls have previously been placed on fuel stored in Pools C and D. These changes will eliminate the dependence on Boraflex in the PWR ''flux trap'' style storage racks. These changes do not impact the probability of having a fuel handling accident and do not impact the consequences of a fuel handling accident.

Therefore, this amendment 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.

No change is being made to the acceptance criteria of the dry storage of new fuel. These criteria are being added to Technical Specification Section 5.6.1. Detailed analyses have been performed to ensure a criticality accident in Pools ''A'' and ''B'' is not a credible event. The events that could lead to a criticality accident are not new. These events include a fuel mis-positioning event, a fuel drop event, and a boron dilution event. The proposed changes do not impact the probability of any of these events. The detailed criticality analyses performed demonstrate that criticality would not occur following any of these events. For the more likely event, such as a fuel mis-positioning event, the acceptance criteria for k_{eff} remains less than or equal to 0.95. For the unlikely event that the spent fuel storage pool boron concentration was reduced to zero, k_{eff} remains less than 1.0.

Therefore, a criticality accident remains ''not credible,'' and this amendment 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.

Incorporation of acceptance criteria for dry storage of new fuel into TS 5.6.1 does not involve a reduction in the margin of safety. The new fuel storage condition continues to meet $k_{eff} \leq 0.95$ during normal conditions and $k_{eff} \leq 0.98$ under optimal moderation conditions.

The proposed changes for storage of new and irradiated fuel in Pools ''A'' and ''B'' continue to provide the controls necessary to ensure a criticality event could not occur in the spent fuel storage spool. The acceptance criteria are consistent with the acceptance criteria specified in 10 CFR 50.68, which provide an acceptable margin of safety with regard to the potential for a criticality event.

Therefore, this 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: September 26, 2005.

Description of amendment request: The proposed amendment will revise the analysis method used for the large-break loss-of-coolant accident (LBLOCA) by incorporating the use of a new approach (ASTRUM) for the treatment of parameter uncertainties. The new approach is described in Westinghouse Topical Report WCAP–16009–P–A, approved by the NRC on November 5, 2004.

Changes to the Technical Specifications to reflect the proposed use of ASTRUM in LBLOCA analysis consist of revisions to the list of references provided in Technical Specification Section 5.6.5, Core Operating Limits Report.

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

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

Response: No.

The proposed change modifies the analysis methodology used to account for the variation in parameters that are used for the safety analysis of the LBLOCA. This proposed change has no effect on the design or operation of plant equipment. Use of the new methodology will revise the results of the current analysis, but there will be no change in initiating events for this accident scenario or the ability of the plant equipment or plant operators to respond.

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

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

Response: No.

The proposed change does not involve modifications to existing plant equipment or the installation of any new equipment. The proposed change only affects the analysis methodology that is used to evaluate the response of existing plant equipment to the LBLOCA scenario. Plant operating and emergency procedures that are in place for the LBLOCA scenario are also not being changed by this proposed amendment. This proposed change does not create new failure modes or malfunctions of plant equipment nor is there a new credible failure mechanism.

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 license amendment revises the analysis methodology which is used to assess the impact of the LBLOCA scenario with respect to established acceptance criteria. Margins of safety for LBLOCA include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria and the associated margins of safety are not being changed. The evaluation of the LBLOCA scenario, using the proposed new methodology must still meet the existing established acceptance criteria.

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 Section Chief: Richard J. Laufer.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: April 4, 2005.

Description of amendment request: The proposed amendments would revise the maximum and minimum allowable values for the degraded voltage function of the 4160 volt essential service system (ESS) bus under-voltage 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 proposed changes revise the Technical Specifications (TS) maximum and minimum allowable values for the degraded voltage protection function and implement the use of automatic load tap changers (LTCs) on transformers that provide power to safety-related equipment. The only accident previously evaluated for which the probability is potentially affected by these changes is the loss of offsite power (LOOP). An allowable value for the degraded voltage protection function that is too high could cause the emergency buses to transfer to the emergency diesel generators (EDG) and thus increase the probability of a LOOP. The allowable value for the degraded voltage protection function has been revised in accordance with an NRC-approved setpoint methodology and will continue to ensure that the degraded voltage protection function actuates when required, but does not actuate prematurely to cause a LOOP.

A failure of an LTC while in automatic operation mode that results in decreased voltage to the ESS buses could also cause a LOOP. This could occur in two ways. A failure of the LTC controller that results in rapidly decreasing the voltage to the emergency buses is the most severe failure mode. However, a backup controller is provided with the LTC that makes this failure highly unlikely. A failure of the LTC controller to respond to decreasing grid voltage is less severe, since grid voltage changes occur slowly. In both of the above potential failure modes, operators will take manual control of the LTC to mitigate the effects of the failure. Thus, the probability of a LOOP is not significantly increased.

The proposed changes will have no effect on the consequences of a LOOP, since the EDGs provide power to safety related equipment following a LOOP. The EDGs are not affected by the proposed changes.

The probability of other accidents previously evaluated is not affected, since the proposed changes do not affect the way plant equipment is operated and thus do not contribute to the initiation of any of the previously evaluated accidents. The only way in which the consequences of other previously evaluated accidents could be affected is if a failure of the LTC while in automatic operation mode caused a sustained high voltage which resulted in damage to safety related equipment that is used to mitigate an accident. Damage due to over-voltage is time-dependent. Since the LTC is equipped with a backup controller, and since operator action is available to prevent a sustained high voltage condition from occurring, damage to safety related equipment is extremely unlikely, and thus the consequences of these accidents are not significantly increased.

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

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

accident from any accident previously evaluated?

Response: No.

The proposed changes involve functions that provide offsite power to safety related equipment for accident mitigation. Thus, the proposed changes potentially affect the consequences of previously evaluated accidents (as addressed in Question 1), but do not result in any new mechanisms that could initiate damage to the reactor and its principal safety barriers (i.e., fuel cladding, reactor coolant system, or primary containment).

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 do not affect the inputs or assumptions of any of the analyses that demonstrate the integrity of the fuel cladding, reactor coolant system, or containment during accident conditions. The allowable values for the degraded voltage protection function have been revised in accordance with an NRC-approved setpoint methodology and will continue to ensure that the degraded voltage protection function actuates when required, but does not actuate prematurely to cause a LOOP. Automatic operation of the LTC increases margin by reducing the potential for transferring to the EDGs during an event.

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 requested amendments involve no significant hazards consideration.

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

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: September 22, 2005.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 operating license and Technical Specifications to increase the licensed rated power level by 1.7 percent from 3587 megawatts thermal (MWt) to 3648 MWt. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Seabrook Station performed evaluations of the Nuclear Steam Supply System (NSSS) and balance of plant systems, components, and analyses that could be affected by the proposed change. A power uncertainty calculation was performed, and the effect of increase core thermal power by 1.7 percent to 3648 MWt on the Seabrook Station design and licensing basis was evaluated. The result of the evaluations determined that all systems and components continue to be capable of performing their design function at the MUR [measurement uncertainty recapture] core power level of 3648 MWt. An evaluation of the accident analyses demonstrates that the applicable analyses acceptance criteria continue to be met. No accident initiators are affected by the MUR power uprate and no challenges to any plant safety barriers are created by the proposed change.

The proposed change does not affect the release paths, the frequency of release, or the analyzed source term for any accidents previously evaluated in the Seabrook Station Updated Final Safety Analysis Report (UFSAR). Systems, structures, and components required to mitigate transients continue to be capable of performing their design functions, and thus were found acceptable. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at the MUR core power level of 3648 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source term used to assess radiological consequences [has] been reviewed and determined to bound operation at the MUR core power level.

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

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

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. The installation of the Caldon LEFM CheckPlus™ System has been analyzed, and failures of the system will have no adverse effect on any safety-related system or any systems, structures, and components required for transient mitigation. Systems, structures, and components previously required for the mitigation of a transient continue to be capable of fulfilling their intended design functions. The proposed change has no adverse affect on any safety-related system or component and does not change the performance or integrity of any safety-related system.

The proposed change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 3648 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at the MUR core power level of 3648 MWt. Credible malfunctions continue to be bounded by the current accident analyses of record or evaluations that demonstrate that applicable criteria continue to be met.

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

3. The proposed change will not involve a significant reduction in a margin [of] safety.

The margins of safety associated with the MUR are those pertaining to core thermal power. These include those associated with the fuel cladding, Reactor Coolant System pressure boundary, and containment barriers. An engineering evaluation of the 1.7 percent increase in core thermal power from 3587 MWt to 3648 MWt was performed. The current licensing bases analyzed core power is 3659 MWt. The analyzed core power level of 3659 MWt bounds the NSSS thermal and hydraulic parameters at the MUR core power level of 3648 MWt. The NSSS systems and components were evaluated at the MUR core power level and it was determined that the NSSS systems and components continue to operate satisfactorily at the MUR power level. The NSSS accident analyses were evaluated at the MUR core power level of 3648 MWt. In all cases, the accident analyses at the MUR core power level of 3648 MWt were bounded by the current licensing bases analyzed core power level of 3659 MWt. As such, the margins of safety continue to be bounded by the current analyses of record for this change.

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: M. S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: September 29, 2005.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1, Technical Specifications (TSs) to permit a one-time, six-month extension to the currently approved 15-year test interval for the containment integrated leak rate test.

Basis for proposed no significant hazards consideration determination:

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

1. The proposed change [does] not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability or consequences of accidents previously evaluated in the UFSAR [updated final safety analysis report] are unaffected by this proposed change. There is no change to any equipment response or accident mitigation scenario, and this change results in no additional challenges to fission product barrier integrity. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. As a result, the outcomes of previously evaluated accidents are unaffected. The proposed extension to the containment integrated leak rate test (ILRT) interval does not involve a significant increase in consequences because, as discussed in NUREG 1493, Performance Based Containment Leak Rate Test Program, Type B and C tests identify the vast majority (greater than 95 percent) of all potential leakage paths. Further, ILRTs identify only a few potential leakage paths that cannot be identified through Type B and C testing, and leaks found by Type A testing have been only marginally greater than existing requirements. In addition, periodic inspections ensure that any significant containment degradation will not go undetected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. The proposed change neither installs or removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system, or component. No physical changes are being made to the plant, so no new accident causal mechanisms are being introduced. The proposed change only changes the frequency of performing the ILRT; however, the test implementation and acceptance criteria are unchanged. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change [does] not involve a significant reduction in a margin of safety.

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. The ability of any operable structure, system, or component to perform its designated safety function is unaffected by this change. NUREG 1493 concluded that reducing the frequency of ILRTs to 20 years resulted in an imperceptible increase in risk. Also, inspections of containment, required by the ASME code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] and the maintenance rule, ensure that containment will not degrade in a manner that is only detectable by Type A (ILRT) testing. Therefore, the margin of safety as defined in the TS is not reduced and 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: M. S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: September 29, 2005.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 Technical Specifications to permit a change in the steam generator tube inspection requirements to include a sampling of the bulges and over-expansions for portions of the steam generator tubes within the hot leg tubesheet region.

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

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

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the steam generator inspection criteria do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed changes to the steam generator tube inspection criteria, are the steam generator tube rupture (SGTR) event and the steam line break (SLB) accident.

During the SGTR event, the required structural integrity margins of the steam generator tubes will be maintained by the presence of the steam generator tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes,'' are maintained for both normal and postulated accident conditions.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated ruptured tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically-expanded outside diameter.

Furthermore, the proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

The probability of a[n] SLB accident is unaffected by the potential failure of a steam generator tube as this failure is not an initiator for a[n] SLB accident.

The consequences of a[n] SLB accident are also not significantly affected by the proposed changes. During a[n] SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a[n] SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the

observation that while the driving pressure causing leakage increases by approximately a factor of (two) 2, the flow resistance associated with an increase in tube-to-tubesheet contact pressure, during a[n] SLB accident, increases by approximately a factor of 2.5. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate even if the increase in contact pressure is ignored. Since normal operating leakage (spiking) is limited to less that 0.104 gpm (150 gpd) for continued power operation per station operating procedure OS 1227.02, ''Steam Generator Tube Leak,'' the associated accident condition leak rate, assuming all leakage to be from lower tube sheet indications, would be bound by 0.208 gpm (twice normal operating leak rate). This value is well within the assumed accident leakage rate of 0.347 gpm discussed in the Seabrook Station Updated Safety Analysis Report, Section 15.1.5 ''Steam System Piping Failure.'' Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges / overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a[n] SLB accident remain unaffected.

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

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

The proposed changes do not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' and NRC Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, ''Reactor Coolant Pressure Boundary,'' GDC 15, ''Reactor Coolant System Design,'' GDC 31, ''Fracture Prevention of Reactor Coolant Pressure Boundary,'' and GDC 32, ''Inspection of Reactor Coolant Pressure Boundary,'' by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation the probability and consequences of a SGTR are reduced. RG 1.121 uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse evaluation LTR-CDME–05–170, ''Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Seabrook Generating Station,'' defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited hot leg tubesheet inspection depth criteria.

Therefore, the proposed changes do not involve a significant reduction in any 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: M. S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: September 27, 2005.

Description of amendment request: The amendments proposed by Southern Nuclear Operating Company would revise the Technical Specifications (TS) to eliminate the Power Range Neutron Flux-High Negative Rate Reactor Trip function, based on the approved methodology contained in Westinghouse Topical Report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event.'' The changes will allow the elimination of a trip circuitry that is not credited in the Farley Nuclear Plant safety analysis, and which can result in an unnecessary reactor trip. These changes will be implemented sequentially, concurrent with each unit's refueling outage during which the design change is implemented. Additionally, this amendment request deletes TS Bases text associated with an unconservative local Departure from Nucleate Boiling Ratio.

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?

No. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). All of the safety analyses have been evaluated for impact due to this change. The elimination of the Power Range Neutron Flux-High Negative Rate Reactor Trip function and the elimination of text in the TS [Technical Specifications] Bases for LC0 3.3.1, page B 3.3.1–1 1, associated with an unconservative local DNBR [departure from nucleate boiling ratio], does not affect the dropped RCCA [Rod Cluster Control Assembly] analyses nor any other analyses, since it is not credited in any of the safety analyses; therefore, the probability of an accident has not been increased. All dose consequences have been evaluated with respect to the proposed changes, there is no impact due to the proposed change, and all acceptance criteria continue to be met. Therefore, these 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 previously evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any accident already evaluated in the UFSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as result of the proposed changes. The changes have no adverse effects on any safety-related system. Therefore, all accident analyses criteria continue to be met and these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed changes do not involve a significant reduction in a margin of safety. The dropped RCCA(s) event does not credit the Power Range Neutron Flux-High Negative Rate Reactor Trip function. The conclusion presented in the UFSAR Section 15.2.3.3 that the DNBR design basis is met for a dropped RCCA(s) event remains valid for the proposed changes, which are based on the NRC approved methodology contained in CAP–11394–PA. Additionally, WCAP–11394–P–A indicates that the analysis for a dropped rod event envelops a multiple rod drop accident at high power levels, and that such an accident will not result in an unconservative local DNBR. All applicable acceptance criteria continue to be met. 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Evangelos C. Marinos.

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

Date of amendment request: October 6, 2005.

Description of amendment request: The amendments proposed by Southern Nuclear Operating Company (SNC) would revise the Technical Specifications (TS) to support a revision to the Best Estimate Loss of Coolant Accident (BELOCA) for Farley Nuclear Plant (FNP). The NRC recently approved a new Westinghouse BELOCA methodology, Automated Statistical Treatment of Uncertainty Method (ASTRUM). ASTRUM was submitted in WCAP–16009–P. The NRC issued a Safety Evaluation Report in a letter dated November 5, 2004. Westinghouse issued WCAP–16009–P–A in January 2005. SNC has completed the analysis for FNP and the enclosed proposed amendment is to incorporate a reference to WCAP–16009–P–A in TS section 5.6.5 Core Operating Limits Report (COLR).

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 physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA [Loss of Coolant Accident] (BELOCA) analysis methodology. The proposed TS changes simply involve updating the references in TS 5.6.5.b, Core Operating Limits Report (COLR), to reference the Westinghouse BELOCA analysis methodology. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant; therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, ''Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.'' No other accident consequence is potentially affected by this change.

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

Therefore, it is concluded that this change does not significantly 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 accident previously evaluated?

There are no physical changes being made to the plant as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. No new modes of plant operation are being introduced. The configuration, operation and accident response of the structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the Westinghouse Best Estimate Large Break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any RPS or ESF actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes.

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

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

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

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

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Evangelos C. Marino.

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: January 27, 2005.

Description of amendment request: The proposed amendments would revise Technical Specifications Limiting Conditions for Operations 3.3.1, 3.3.2, 3.3.6, and 3.3.8, by extending the Surveillance Test Intervals for the Reactor Protection 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. Does the Proposed Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The proposed changes to the Completion Time, bypass test time, and Surveillance Frequencies reduce the potential for inadvertent reactor trips and spurious actuations and, therefore, do not increase the probability of any accident previously evaluated. The proposed changes to the allowed Completion Time, bypass test time, and Surveillance Frequencies do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the reactor trip system and engineered safety feature actuation system (RTS and ESFAS) signals. The RTS and ESFAS will remain highly reliable, and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by core damage frequency (CDF) is less than 1.01E–06 per year and the impact on large early release frequency (LERF) is less than 1.0E–07 per year. In addition, for the Completion Time change, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–08. These changes meet the

acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by CDF, LERF, ICCDP, and ICLERP is within the acceptance criteria of existing regulatory guidance, there will not be a significant increase in the consequences of any accidents. 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 the safety analysis assumptions and resultant consequences. Therefore, it is concluded that this change does not increase the probability of occurrence of a malfunction of equipment important to safety.

2. Does the Proposed Change Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated?

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 setpoints after the proposed changes are implemented. There are no design changes associated with the license amendment. The changes to Completion Time, bypass test time, and Surveillance Frequency do not change any existing accident scenarios, nor create any new or different accident scenarios. 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 possibility of a new or different malfunction of safety related equipment is not created.

3. Does the Proposed Change Involve a Significant Reduction in the 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. 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. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Time, bypass test time, and Surveillance Frequencies, it is expected there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing. 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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Evangelos C. Marinos.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant (SQN), Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: September 30, 2005 (TS–05–02).

Description of amendment request: The proposed amendment would revise the SQN Technical Specification (TS) Section 5.0, ''Design Features,'' to more conform with NUREG–1431 Revision 3, ''Standard Technical Specifications for Westinghouse Plants.'' The proposed change included the elimination of exclusion area, low population zone, and effluent subsections and associated figures referred to in Section 5.1, ''Site''; elimination of Section 5.2, ''Containment''; elimination of Section 5.4, ''Reactor Coolant System,'' as well as Section 5.5, ''Meteorological Tower Location,'' and its figure. Lastly, a proposed change to the TS ''Administrative Control'' section to acquire the component cyclic or transient limits currently located in the ''Design Features'' section.

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 removal of information and figures featuring the locations of the site exclusion area, gaseous and liquid effluent boundaries, low population zone, and the meteorological tower is administrative in nature. Most, if not, all of this information is located in other licensee control documents, such as the Final Safety Analysis Report (FSAR). Congruently, the addition of a site location description only adds geographical information to the TSs. The relocation and revision of the component cyclic or transient limits requirement does not alter the requirement to track and maintain these limits and thus considered administrative. This proposed amendment involves no technical changes to the existing TSs and does not impact initiators of analyzed events. The changes also do not impact the assumed mitigation of accidents or transient events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a change to plant systems, components, or operating practices that could result in a change in accident generation potential. The proposed changes do not impose any new or different requirements or eliminate any existing requirements. The proposed changes do not alter assumptions made in the safety analyses and licensing basis. 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 deletion of information and figures featuring the locations of the site exclusion area, gaseous and liquid effluent boundaries, low population zone, and the meteorological tower does not affect operational limits or functional capabilities of plant systems, structures and components. The addition of a site location description adds geographical information to the TSs. The relocation and revision of the component cyclic or transient limits requirements also does not affect operational limits or functional capabilities of plant systems, structures and components. These changes pose no effect on margin of safety. The TS identified maximum steel containment temperature value is not the current limiting design value, which is found in the FSAR. Its elimination is considered administrative in nature and does not result in a change of margin of safety to the containment design. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three

standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: April 27, 2005, as supplemented by letter dated July 20, 2005.

Brief description of amendments: The amendment revises Technical Specification (TS) 5.6.5, ''Core Operating Limits Report,'' by adding topical report WCAP–13060–P–A, ''Westinghouse Fuel Assembly Reconstitution Evaluation Methodology,'' to the list of approved methodologies to be used at Comanche Peak Steam Electric Station (CPSES), Unit 2.

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 change is administrative in nature and as such does not impact the condition or performance of any plant structure, system or component. The core operating limits are established to support Technical Specifications 3.1, 3.2, 3.3, 3.4, and 3.9. The core operating limits ensure that fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). The methods used to determine the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in TS section 5.6.5.b. Application of these approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and AOOs. The requested Technical Specification change does not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested change does not affect any postulated accident precursors, does not affect any accident mitigation systems, and does not introduce any new accident initiation mechanisms.

As a result, the proposed change to the CPSES Technical Specifications does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed administrative change.

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 change is administrative in nature, and therefore does not involve any change in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions, and no new failure modes are being introduced.

Therefore, the proposed administrative change to the CPSES Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any accident previously evaluated.

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

Response: No.

The proposed change is administrative in nature and does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the proposed administrative change.

Therefore, the proposed change to the CPSES Technical Specifications does not involve any 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 Section 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: January 24, 2005.

Brief description of amendments: The amendments will revise the surveillance requirements (SRs) for Technical Specification 3.7.5, ''Auxilary Feed Water (AFW) System.'' Specifically, a note will be added to SRs 3.7.5.1, 3.7.5.3, and 3.7.5.4 that states, ''AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.''

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 change has no impact on the probability of any accident previously evaluated. The consequences of the limiting transients and accidents (full power operation) are unaffected by the proposed change. At low power sufficient time is available to establish auxiliary feedwater injection if needed.

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.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. There are no changes in the method by which any safety-related plant system performs its safety function. Overall protection system performance will remain within the bounds of the previously performed accident analyses and the protection systems will continue to function in a manner consistent with the plant design basis. The proposed changes do not affect the probability of any event initiators. The proposed changes do not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the Final Safety Analysis Report (FSAR).

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

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

Response: No.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, the Departure from Nucleate Boiling Ratio (DNBR) limits, the Heat Flux Hot Channel Factor (FQ), the Nuclear Enthalpy Rise Hot Channel Factor (F'H), the Loss of Coolant Accident Peak Centerline Temperature (LOCA PCT), peak local power density, or any other margin of safety. The

radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met. Since the limiting transients and accidents are unaffected, the proposed change[s] do 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 Section 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: August 10, 2005.

Brief description of amendments: The amendments would revise the Technical Specification (TS) 5.5.13, ''Diesel Fuel Oil Testing Program,'' to relocate the specific American Society for Testing and Materials (ASTM) Standard reference from the Administrative Controls Section of TS to a licensee-controlled document.

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 relocate the specific American Society for Testing and Materials (ASTM) Standard references from the Administrative Controls of TS to a licensee-controlled document. Since any change to the licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, tests and experiments,'' no increase in the probability or consequences of an accident previously evaluated is involved.

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 individual or cumulative occupational or public radiation exposure.

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 do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or change in the methods governing normal plant operation. In addition, the changes do not alter the assumptions made in the analysis and licensing basis.

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

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

Response: No.

The level of safety of facility operation is unaffected by the proposed changes since there is no change in the intent of the TS requirements of assuring fuel oil is of the appropriate quality for emergency DG [diesel generator] use. The proposed changes provide the flexibility needed to utilize state-of-the-art technology in fuel oil sampling and analysis methods.

Therefore the proposed changes do 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 Section 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: August 22, 2005.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.7.10, ''Control Room Emergency Filtration/Pressurization System (CREFS) and Control Room Envelope (CRE),'' and adds new TS 5.5.20, ''Control Room Integrity Program,'' and TS 5.6.11, ''Control Room Report.'' In addition the amendments update the Final Safety Analysis Report to include new methods and assumptions as described in Regulatory Guide 1.195 for evaluation of radiological consequences.

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 change addresses the Control Room Envelope (CRE), including updated surveillances for the Control Room Emergency Filtration/Pressurization System (CREFS) trains and the CRE, a new TS 5.5.20, ''Control Room Integrity Program,'' and a new TS 5.6.11, ''Control Room Report.'' These changes are consistent with the guidance in Regulatory Guides 1.196 and 1.197. New methods and assumptions for evaluating radiological consequences for design basis accidents are adopted consistent with NRC Regulatory Guide 1.195. The acceptance limits for the Control Room Integrity Program are based on these revised radiological dose consequences calculations. The proposed changes do not adversely affect accident initiators or precursors nor alter the configuration of the facility. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event to within the Regulatory Guide 1.195 acceptance limits. This activity is a revision to the Technical Specifications and the supporting radiological dose consequences analyses for the control room ventilation system which is a mitigating system designed to minimize in-leakage into the control room and to filter the control room atmosphere to protect the control room operators following accidents previously analyzed. An important part of the system is the control room envelope (CRE). The CRE integrity is not an initiator or precursor to any accident previously evaluated. Therefore the probability of occurrence of any accident previously evaluated is not increased. Performing tests and implementing programs that verify the integrity of the CRE and control room habitability ensure mitigation features are capable of performing the assumed function.

The revised radiological consequences analyses, performed using the assumptions and methodologies presented in Regulatory Guidance 1.195, do not result in significant increases in the radiological dose consequences to the general public nor to the control room operators. All calculated dose consequences are within acceptance limits of Regulatory Guide 1.195.

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 will not alter the requirements of the control room ventilation

system or its function during accident conditions. No new or different accidents result from performing the new revised actions and surveillances or programs required. 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 which could create the possibility of a new or different kind of accident. The proposed changes are consistent with the safety analysis assumptions and current plant operating practices. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The 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 affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without mitigating actions. The proposed changes do not affect systems that are required to respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

Therefore the proposed change does not involve a 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 Section Chief: David Terao.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: May 27, 2005.

Brief description of amendment: The amendment revised the technical specification (TS) testing frequency for the surveillance requirement (SR) in TS 3.1.4, ''Control Rod Scram Times.'' Specifically, the change revised the frequency for SR 3.1.4.2, ''Control Rod Scram Time Testing,'' from ''120 days cumulative operation in MODE 1'' to ''200 days cumulative operation in MODE 1.''

Date of issuance: October 25, 2005.

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

Amendment No.: 167.

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

*Date of initial notice in **Federal Register:*** July 19, 2005 (70 FR 41443).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 31, 2005.

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

Date of issuance: October 20, 2005.

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

Amendment No.: 284.

Facility Operating License No. DPR–59: The amendment revised the TSs.

*Date of initial notice in **Federal Register:*** August 16, 2005 (70 FR 48204).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 17, 2004, as supplemented by letter dated September 28, 2005.

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

Date of issuance: October 18, 2005.

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

Amendment No.: 145.

Facility Operating License Nos. NPF–37 and NPF–66: The amendments revised the Environmental Protection Plan.

*Date of initial notice in **Federal Register:*** April 12, 2005 (70 FR 19115). The supplement dated September 28, 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 October 18, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 25, 2004, as supplement by letter dated August 1, 2005.

Brief description of amendment: The amendment revises the required channels per trip system for several instrument functions contained in Technical Specification Tables 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' 3.3.6.2–1, ''Secondary Containment Isolation Instrumentation,'' and 3.3.7.1–1 ''Control Room Emergency Filter System Instrumentation.''

Date of issuance: October 27, 2005.

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

Amendment No.: 212.

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

Date of initial notice in ***Federal Register:*** January 4, 2005 (70 FR 402).

The supplement dated August 1, 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 October 27, 2005.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–272, Salem Nuclear Generating Station Unit No. 1, Salem County, New Jersey

Date of application for amendment: February 23, 2005, as supplemented by letters dated August 2, 2005, and September 21, 2005.

Brief description of amendment: The amendments revised Technical Specifications (TSs) to implement a new steam generator tube surveillance program that is consistent with the program proposed by the TS Task Force (TSTF) in TSTF–449.

Date of issuance: October 14, 2005.

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

Amendment No.: 268.

Facility Operating License No. DPR–70: The amendments revised the TSs.

Date of initial notice in ***Federal Register:*** May 10, 2005 (70 FR 24655). Supplements dated August 2, 2005, and September 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 amendments is contained in a Safety Evaluation dated October 14, 2005.

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: March 4, 2005, as supplemented August 2, 2005.

Brief description of amendments: These amendments extend the completion time from 1 hour to 24 hours for Actions ''a'' and ''b'' of Salem Nuclear Generating Station, Unit Nos. 1 and 2 Technical Specification (TS) 3.5.1, ''Accumulators,'' which requires restoration of an accumulator when it has been declared inoperable for reasons other than boron concentration in the accumulator not being within the required range.

Date of issuance: October 14, 2005.

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

Amendment Nos.: 267 and 249.

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

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29800). The August 2, 2005, supplement provided clarifying information only and did not change the scope of the proposed amendment, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 14, 2005.

No significant hazards consideration comments received: No.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: January 24, 2005.

Brief description of amendment: The amendment removes unnecessary and obsolete information from the facility operating license.

Date of issuance: September 21, 2005.

Effective date: September 21, 2005.

Amendment No.: 132.

Facility Operating License No. DPR–54: The amendment revised the License.

Date of initial notice in ***Federal Register:*** March 29, 2005 (70 FR 15947).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 2005.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 12, 2004.

Brief description of amendments: The amendments revised Surveillance Requirement (SR) 4.7.8.d.3 of the Auxiliary Building Gas Treatment System (ABGTS) by deleting vacuum relief flow requirements. The change removes criteria from the SR that is not necessary to verify the operability of the ABGTS and eliminates confusion regarding the basis for the vacuum relief flow requirement.

Date of issuance: August 18, 2005.

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

Amendment Nos.: 303 and 293.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the technical specifications.

Date of initial notice in ***Federal Register:*** October 12, 2004 (69 FR 60687).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 18, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 27, 2004, as supplemented by letter dated June 17, 2005.

Brief description of amendment: The amendment (1) deleted Conditions 2.C.(3), 2.C.(4), 2.C.(6) through 2.C.(14), Section 2.F, and Attachments 1 and 2, and (2) revised Conditions 2.C.(1) and 2.C.(5), to the facility operating license, to reflect completed requirements. In addition, the list of attachments and appendices to the operating license was revised to reflect the deletion of Attachments 1 and 2. The proposed

changes to Technical Specifications Table 5.5.9–2, ''Steam Generator Tube Inspection,'' and Table 5.5.9–3, ''Steam Generator Repaired Tube Inspection,'' were also submitted in the licensee's application dated September 17, 2004 (ULNRC–05056), for the replacement steam generator project and were approved in Amendment No. 168, which was issued in the NRC letter dated September 29, 2005.

Date of issuance: October 25, 2005.

Effective date: October 25, 2005, and shall be implemented within 90 days of the date of issuance.

Amendment No.: 169.

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

Date of initial notice in ***Federal Register:*** December 7, 2004 (69 FR 70723). The June 17, 2005, supplemental 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 no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2005.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 31st day of October, 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 05–22002 Filed 11–7–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Interim Staff Guidance Documents for Fuel Cycle Facilities

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

FOR FURTHER INFORMATION CONTACT: James Smith, Project manager, Technical Support Group, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20005–0001. Telephone: (301) 415–6459; fax number: (301) 415–5370; e-mail: *jas4@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

The Nuclear Regulatory Commission (NRC) continues to prepare and issue Interim Staff Guidance (ISG) documents for fuel cycle facilities. These ISG documents provide clarifying guidance to the NRC staff when reviewing licensee integrated safety analysis, license applications or amendment requests or other related licensing activities for fuel cycle facilities under subpart H of 10 CFR part 70. FCSS–ISG–08 has been issued and is provided for information.

II. Summary

The purpose of this notice is to provide notice to the public of the issuance of FCSS–ISG–08, Revision 0, which provides guidance to NRC staff to address accident sequences that may result from natural phenomena hazards relative to license application or amendment request under 10 CFR Part 70, Subpart H. FCSS–ISG–08, Revision 0, has been approved and issued after a general revision based on NRC staff and public comments on the initial draft.

III. Further Information

The document related to this action is available electronically at the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* From this site, you can access the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The ADAMS ascension number for the document related to this notice is provided in the following table. If you do not have access to ADAMS or if there are problems in accessing the document located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Interim staff guidance	ADAMS Accession No.
FCSS Interim Staff Guidance-08, Revision 0.	ML052650305

This document may also be viewed electronically on the public computers located at the NRC's PDR, O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee. Comments on these documents may be forwarded to James Smith, Project Manager, Technical Support Group, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20005–0001. Comments can also be submitted by telephone, fax, or e-mail which are as follows: Telephone: (301) 415–6459; fax number: (301) 415–5370; e-mail: *jas4@nrc.gov.*

Dated at Rockville, Maryland this 27th day of October 2005.

For the Nuclear Regulatory Commission.

Melanie A. Galloway,

Chief, Technical Support Group, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

Attachment—FCSS Interim Staff Guidance-08, Revision 0, Natural Phenomena Hazards

Prepared by Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards

Issue

Additional guidance is required to address accident sequences that may result from natural phenomena hazards in the context of a license application or an amendment request under Title 10 Code of Federal Regulations (10 CFR) part 70, subpart H.

Introduction

This Interim Staff Guidance (ISG) provides additional guidance for reviewing the applicant's (or licensee's) evaluation of natural phenomena hazards up to and including ''highly unlikely'' events for both new and existing facilities.

Discussion

The performance requirements of 10 CFR 70.61 for facilities processing special nuclear materials require that individual accident sequences resulting in high consequences to workers and the public be ''highly unlikely'' and that sequences resulting in intermediate consequences to these receptors be ''unlikely.'' Although the threshold levels that differentiate high consequence events from intermediate consequence events are established in the regulations, the definitions of ''highly unlikely'' and ''unlikely'' are not. Definitions of these terms must be described in the integrated safety analysis (ISA) summary submitted by applicants and licensees according to 10 CFR 70.65(b)(9) and subjected to staff approval. Further description of the acceptance criteria for the definitions of these terms can be found in Chapter 3 of NUREG–1520, ''Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility.''

The implementation of these requirements may vary somewhat due to different definitions of likelihood proposed by different applicants (or