7590-01-P

UNITED STATES 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 September 25, 2008 to October 8, 2008. The last biweekly notice was published on October 7, 2008 (73 FR 58669).

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.

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

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and

page number of this *Federal Register* notice. Written comments may also be delivered to Room 6D44, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, person(s) 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 via electronic submission through the NRC E-Filing system 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 person(s) 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 hearing or a petition for leave to intervene must be filed in accordance with the NRC E-Filing rule, which the NRC promulgated in August 28, 2007 (72 FR 49139). The E-Filing process requires participants to submit and serve documents over the internet or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least five (5) days prior to the filing deadline, the petitioner/requestor must contact the Office of the Secretary by e-mail at hearingdocket@nrc.gov, or by calling (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and/or (2) creation of an electronic docket for the proceeding (even in instances in which the petitioner/requestor

(or its counsel or representative) already holds an NRC-issued digital ID certificate). Each petitioner/requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate is available on NRC's public website at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.

Once a petitioner/requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, it can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public website at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the filer submits its documents through EIE. To be timely, an electronic filing must be submitted to the EIE system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document.

The EIE system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the "Contact Us" link located on the NRC website at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC

technical help line, which is available between 8:30 a.m. and 4:15 p.m., Eastern Time, Monday through Friday. The help line number is (800) 397-4209 or locally, (301) 415-4737.

Participants who believe that they have a good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted 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; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville, Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Non-timely requests and/or petitions and contentions will not be entertained absent a determination by the Commission, the presiding officer, or the Atomic Safety and Licensing Board that the petition and/or request should be granted and/or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)-(viii). To be timely, filings must be submitted no later than 11:59 p.m. Eastern Time on the due date.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at http://ehd.nrc.gov/EHD_Proceeding/home.asp, unless excluded pursuant to an order of the Commission, an Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their

filings. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

For further details with respect to this amendment 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 email to pdr.resource@nrc.gov.

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

Date of amendments request: August 27, 2008

Description of amendments request: The amendment would change the containment buffering agent from trisodium phosphate (TSP) to sodium tetraborate in order to minimize the potential for sump screen blockage due to potential adverse chemical interactions between TSP and certain insulation materials used in containment under post loss-of-coolant accident conditions. This amendment is one of the remaining modifications required for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 to achieve full compliance with the requirements of Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (Agencywide Documents Access and Management System (ADAMS) Accession Number ML042360586).

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 amendment does not involve a significant increase in the probability of an accident previously evaluated because the containment buffering agent is not an initiator of any analyzed accident. The proposed change does not impact any failure modes that could lead to an accident. The proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated. The buffering agent in Containment is designed to buffer the acids expected to be produced after a loss-of-coolant accident (LOCA) and is credited in the radiological analysis for iodine retention. Utilizing the required quantity of sodium tetraborate decahydrate (STB) as a buffering agent ensures the post-LOCA containment sump mixture will have a pH ≥ 7.0. The proposed change of replacing trisodium phosphate (TSP) with STB results in the radiological consequences remaining within the limits of 10 CFR 50.67. There is no dose change with the pH ≥ 7.0.

   Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously 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 amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The STB is a passive component that is proposed to be used as a buffering agent to increase the pH of the initially acidic post-LOCA containment water to a more neutral pH. Changing the proposed buffering agent from TSP to STB does not constitute an accident initiator or create a new or different kind of accident than previously analyzed. The proposed amendment does not involve operation of any required systems, structures, or components in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the changes being requested. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response – No

The proposed amendment does not involve a significant reduction in a margin of safety. The proposed amendment of changing the buffering agent from TSP to STB results in equivalent control of maintaining sump pH at ≥ 7.0, thereby controlling containment atmosphere iodine and ensuring the radiological consequences of a LOCA are within regulatory limits. The change of buffering agent from TSP to STB also reduces the amount of calcium phosphate precipitate generated thereby reducing the overall amount of precipitate that may be formed in a postulated LOCA. The buffer change would minimize the potential chemical effects and should enhance the ability of the Emergency Core Cooling System to perform the post-LOCA mitigating functions.

Therefore, the proposed amendment 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 amendments request involves no significant hazards consideration.

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

NRC Branch Chief: Mark G. Kowal

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: July 21, 2008

Description of amendment request: The amendment proposes a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to support adoption of Technical Specification Task Force (TSTF) 359, “Increased Flexibility in Mode Restraints.” The NRC approved adoption of TSTF-359 for ANO-1 in TS Amendment 232. The overall intent of TSTF-359 was to eliminate exceptions to Limiting Condition for Operation (LCO) 3.0.4 within individual specifications and provide requirements within LCO 3.0.4 to control mode changes when TS-

required equipment is inoperable. Following implementation of TS Amendment 232, Entergy discovered that one of the marked-up TS pages which contained an LCO 3.0.4 exception was not provided to the NRC for review in the original submittal.

The NRC staff issued a notice of opportunity for comment in the Federal Register on August 2, 2002 (67 FR 50475), as part of the Consolidated Line Item Improvement Process (CLIIP), on possible amendments to revise the plant-specific TS to modify requirements for model change limitations in LCO 3.0.4 and SR 3.0.4.

The NRC staff subsequently issued a notice of availability of the models for Safety Evaluation and No Significant Hazards Consideration Determination for referencing in license amendment applications in the Federal Register on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the CLIIP, including the model No Significant Hazards Consideration Determination, in its application dated October 22, 2007.

The proposed TS changes are consistent with NRC-approved Industry TSTF STS change, TSTF-359, Revision 8, as modified by 68 FR 16579. TSTF-359, Revision 8, was subsequently revised to incorporate the modifications discussed in the April 4, 2003, Federal Register notice and other minor changes. TSTF-359, Revision 9, was subsequently submitted to the NRC on April 28, 2003, and was approved by the NRC on May 9, 2003.

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

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

   Response: No

   The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required

actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

   Response: No

   The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS Limiting Conditions for Operation (LCO). The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing

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

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

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

NRC Branch Chief: Michael T. Markley

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

Date of amendment request: December 13, 2007

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) Section 4.3.1, "Criticality," to add a new requirement to use a blocking device in spent fuel storage rack cells that cannot maintain the effective neutron multiplication factor, K_{eff}, requirements specified in TS Section 4.3.1.1.a. In addition, the proposed change revises TS Section 4.3.3 to reflect that the LaSalle County Station, Unit 2 spent fuel storage capacity is limited to no more than a combination of 4078 fuel assemblies and blocking devices.

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

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

   Response: No

   The proposed change adds an additional requirement to the TS to ensure that the effective neutron multiplication factor K_{eff}, is less than or equal to 0.95, if fully flooded with borated water. The additional requirement is to insert a blocking device into unusable storage rack cell locations. Since the proposed change

pertains only to the spent fuel pool (SFP), only those accidents that are related to movement and storage of fuel assemblies in the SFP could be potentially affected by the proposed change.

The probability that a misplaced fuel assembly would result in an inadvertent criticality is unchanged since the process and procedural controls governing fuel cell movement in the SFP will not be changed. The current criticality analysis for the LSCS Unit 2 SFP credits the neutron absorbing properties of the Boraflex neutron poison material in the spent fuel storage racks. The current analysis demonstrates: (1) adequate margin to criticality for all spent fuel storage cells, (2) adequate margin for fuel assemblies inadvertently placed into locations adjacent to the spent fuel racks, and (3) adequate margin for assemblies accidentally dropped onto the spent fuel racks. The dose consequences of the most limiting drop of a fuel assembly in the spent fuel pool is limited by the number of the fuel rods damaged and other engineered features unaffected by the proposed change, including the fuel design, fuel decay time, water level in the spent fuel pool, water temperature of the spent fuel pool, and the engineering features of the Reactor Building Ventilation System.

The revised analysis does not result in a significant increase in the probability of an accident previously analyzed. The revised analysis takes no credit for the Boraflex material. The use of a blocking device prevents an inadvertent action to insert a spent fuel assembly, and prevents an assembly that is accidentally dropped to penetrate into the empty spent fuel cell. In addition to this blocking device, administrative controls will be implemented to prevent insertion of a bundle into a cell that is blocked. The probability that a fuel assembly would be inadvertently placed into a location adjacent to the racks is unchanged, and the probability that a fuel assembly would be dropped is unchanged by the revised analysis. These events involve failures of administrative controls, human performance, and equipment failures that are unaffected by the presence or absence of Boraflex and the blocking devices.

The revised analysis does not result in a significant increase in the consequence of an accident previously analyzed. The revised analysis demonstrates adequate margin to criticality for unblocked cells in the LSCS Unit 2 SFP, adequate margin for assemblies inadvertently placed into locations adjacent to the spent fuel racks, and adequate margin for assemblies accidentally dropped onto the spent fuel racks. Placing a spent fuel assembly into a location containing a blocking device is not a credible event since there are diverse and redundant administrative and physical barriers to prevent that.

The revised analysis does not affect the consequences of a dropped fuel assembly. The consequences of dropping a fuel assembly onto any other fuel assembly or other structure, other than a blocking device, are unaffected by the change. The consequences of dropping a fuel assembly onto a blocking device are bounded by the event of dropping an assembly onto another assembly, both for criticality and for radiological consequences. For criticality, the blocking device prevents the dropped assembly from entering the blocked cell. For

radiological consequences, the number of rods damaged when a fuel assembly is accidentally dropped onto a blocking device is bounded the by the number of rods damaged by an assembly dropped onto another assembly. The change does not affect the effectiveness of the other engineered design features to limit the offsite dose consequences of the limiting fuel assembly drop accident.

The proposed change to clarify that the capacity of the Unit 2 SFP is limited to no more than a combination of 4078 fuel assemblies and blocking devices does not affect the probability or consequences of an accident previously analyzed because no physical modifications to the storage racks are proposed. The proposed change will reduce the number of allowable fuel assembly storage locations.

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

Response: No

Onsite storage of spent fuel assemblies in the SFP is a normal activity for which LSCS has been designed and licensed. As part of assuring that this normal activity can be performed without endangering public health and safety, the ability to safely accommodate different possible accidents in the SFP, such as dropping a fuel assembly or misloading a fuel assembly, have been analyzed. The proposed fuel storage configuration does not change the methods of fuel movement or fuel storage. No structural or mechanical change to the racks or fuel handling equipment is being proposed. The proposed change allows for partial use of storage rack locations that have been determined unusable based on the existing criticality analysis.

The blocking devices are passive devices. These devices, when inside a spent fuel storage rack cell, perform the same function of a spent fuel assembly in that cell. These devices do not add any limiting structural loads or affect the removal of decay heat from the other assemblies. The devices are resistant to corrosion and will maintain their structural integrity over the life of the plant. These devices are not under any structural load during normal operations. They are only challenged by an accidental fuel assembly drop. The existing fuel handling accident, which assumes the drop of a fuel bundle, bounds the drop of a blocking device.

This change does not create the possibility of a misloaded assembly into a blocked cell. Placing a spent fuel assembly into a location containing a blocking device is not a credible event since there are diverse and redundant administrative and physical barriers to prevent that.

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

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

Response: No

LSCS TS 4 .3.1 .1 requires the spent fuel storage racks to maintain the effective neutron multiplication factor, K_{eff}, less than or equal to 0.95 when fully flooded with unborated water, which includes an allowance for uncertainties. Therefore, for criticality, the required safety margin is 5% including a conservative margin to account for engineering uncertainties.

The proposed change adds a requirement to use a blocking device to ensure that K_{eff} continues to be less than or equal to 0.95; thus, the required safety margin of 5% is preserved. The proposed change also clarifies that the capacity of the Unit 2 SFP is limited to no more than a combination of 4078 fuel assemblies and blocking devices. This clarification does not impact the required safety margin of 5%.

The current analysis assumes an infinite array of fuel with all fuel at the peak reactivity (i.e., the highest combination of initial enrichment, gadolinium, and fuel burnup that maximizes the reactivity of the fuel). The revised analysis demonstrates the same margin to criticality of 5%, including a conservative margin to account for engineering uncertainties, is maintained assuming an infinite array of fuel with all fuel at the peak reactivity. In addition, the margin of safety for radiological consequences of a dropped fuel assembly are unchanged because the event involving a dropped fuel assembly onto a blocking device is bounded by the consequences of a dropped fuel assembly onto another fuel assembly.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Russell Gibbs

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

Date of amendment request: May 30, 2008, as supplemented on July 17 and September 10, 2008

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) Table 3.3.8.1-1, "Loss of Power Instrumentation," specifically to change the maximum allowable voltage of the 4.16-kV Emergency Bus Undervoltage function from less-than-or-equal to 3899 V to less-than-or-equal-to 3822 V.

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

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

   Response: No.

   The proposed TS change to the maximum allowable voltage for the 4160 volt Emergency Bus Undervoltage relays affects when an Emergency Bus that is experiencing degraded voltage will disconnect from offsite power and transfer to an emergency diesel generator. While the maximum allowed voltage that initiates this action will be lowered, the function remains the same. The maximum allowed voltage has been analyzed to ensure spurious trips will be avoided. The proposed change will not affect any accident initiators or precursors. As a result, the probability of any accident previously evaluated is not significantly increased.

   The consequences of any accident previously evaluated are not increased since the 4160 volt Emergency Bus Undervoltage relays will continue to meet their required function to transfer the 4160 volt Emergency Buses to the emergency diesel generators in the event of a degraded voltage condition on the offsite power supply. This transfer will ensure that the electrical equipment is capable of performing its function to meet the requirements of the accident analyses.

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

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

   Response: No.

No new or different accidents result from utilizing the proposed change. The proposed TS change to the maximum allowable voltage for the 4160 volt Emergency Bus Undervoltage relays does not affect existing or introduce any new accident precursors or modes of operation. The relays will continue to detect undervoltage conditions and transfer the Emergency Buses to the emergency diesel generators at a voltage adequate to ensure proper safety equipment performance and to prevent equipment damage. The function of the relays remains the same.

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

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

Response: No.

The proposed TS change to the maximum allowable voltage for the 4160 volt Emergency Bus Undervoltage relays will allow all safety loads to have sufficient voltage to perform their intended safety functions while ensuring spurious trips are avoided. Thus, the results of the accident analyses will not be affected as the input assumptions are protected.

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

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

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

NRC Branch Chief: Lois M. James

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

Date of amendment request: August 19, 2008

Description of amendment request: The proposed amendment would revise Technical Specification (TS) requirements for mode change limitations in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF-359, Revision 9, "Increase Flexibility in MODE Restraints," and revise TS Section 1.4, "Frequency," in accordance with NRC-approved traveler TSTF-485, Revision 0, "Correct Example 1.4-1."

The NRC staff issued a "Notice of Availability of Model Application Concerning Technical Specification Improvement To Modify Requirements Regarding Mode Change Limitations Using the Consolidated Line Item Improvement Process" in the *Federal Register* on April 4, 2003 (68 FR 16579). The notice referenced a model safety evaluation and a model no significant hazards consideration (NSHC) determination published in the *Federal Register* on August 2, 2002 (67 FR 50475). In its application dated August 19, 2008, the licensee affirmed the applicability of the model NSHC determination which is presented below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC adopted by the licensee regarding TSTF-359 is presented below:

> Criterion 1--The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated
>
> The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> Criterion 2--The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS Limiting Conditions for Operation (LCO). The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

In its application dated August 19, 2008, the licensee also affirmed the applicability of the NSHC approved by the NRC in TSTF-485, 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 revises Section 1.4, Frequency, Example 1.4-1, to be consistent with Surveillance Requirement (SR) 3.0.4 and Limiting Condition for Operation (LCO) 3.0.4. This change is considered administrative in that it modifies the example to demonstrate the proper application of SR 3.0.4 and LCO 3.0.4. The requirements of SR 3.0.4 and LCO 3.0.4 are clear and are clearly explained in the associated Bases. As a result, modifying the example will not result in a change in usage of the Technical Specifications (TS). The proposed change does not adversely affect accident initiators or precursors, the ability of structures, systems, and components (SSCs) to perform their intended function

to mitigate the consequences of an initiating event within the assumed acceptance limits, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Therefore, this change is considered administrative and will have no effect on the probability or consequences of any accident previously evaluated.

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

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

Response: No.

No new or different accidents result from utilizing the proposed change. The change does 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 change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice.

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

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

Response: No.

The proposed change is administrative and will have no effect on the application of the Technical Specification requirements. Therefore, the margin of safety provided by the Technical Specification requirements is unchanged.

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

The NRC staff has reviewed the analysis adopted by the licensee and, based upon this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendment involves NSHC.

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

NRC Branch Chief: Michael T. Markley

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50-220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: August 15, 2008

Description of amendment request: The proposed amendment would revise NMP1 Technical Specification (TS) 6.5.7, "10 CFR 50 [Part 50 of Title 10 of the *Code of Federal Regulations*] Appendix J Testing Program Plan," to allow a one-time extension of the Integrated Leak Rate Test (ILRT) interval for no more than five (5) years. The proposed amendment would allow the next ILRT for NMP1 to be performed within 15 years from the last ILRT as opposed to the current 10-year interval.

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

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

   Response: No.

   The proposed change involves a one-time extension of the primary containment ILRT interval from 10 to 15 years. The proposed change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed change.

   Continued containment integrity is assured by the established programs for local leak rate testing and inservice/containment inspections, which are unaffected by the proposed change. As documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, industry experience has shown that local leak rate tests (Type B and C) have identified the vast

majority of containment leakage paths, and that ILRTs detect only a small fraction of containment leakage pathways.

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 years to 15 years. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be of a magnitude that NUREG-1493 indicates is imperceptible. NMPNS has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed change was determined to be within the guidelines published in NRC Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMPNS has determined that the increase in conditional containment failure probability due to the proposed change would be insignificant. Therefore, it is concluded that the proposed one-time extension of the primary containment ILRT interval from 10 years to 15 years does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change involves a one-time extension of the primary containment ILRT interval. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change in the manner in which the plant is operated or controlled.

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

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

   Response: No.

   The proposed one-time extension of the primary containment ILRT interval does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and

> conditions of the 10 CFR [Part] 50 Appendix J Testing Program Plan, as defined in the TS, exist to ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant safety analyses is maintained. The overall containment leakage rate limit specified by the TS is maintained, and Type B and C containment leakage tests will continue to be performed at the frequency currently required by the TS.
>
> NMP1 and industry experience strongly support the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by the ILRT is small. Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by an ILRT. Additionally, the on-line containment monitoring capability that is inherent to inerted boiling[-]water reactor containments allows for the detection of gross containment leakage that may develop during power operation. This combination of factors ensures that the margin of safety that is inherent in plant safety analyses is maintained. Furthermore, a risk assessment using the current NMP1 Probabilistic Risk Assessment interval events model concluded that extending the ILRT test interval from 10 to 15 years results in a very small change to the NMP1 risk profile.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50-410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: August 14, 2008

Description of amendment request: The proposed amendment would (1) revise the NMP2 Technical Specification (TS) Surveillance Requirement (SR) frequency in TS 3.1.3, “Control Rod

Operability," and (2) revise Example 1.4-3 in TS Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The proposed changes are consistent with Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action." The availability of this TS improvement was announced in the Federal Register on November 13, 2007 (72 FR 63943) as part of the consolidated line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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:

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

The proposed change generically implements TSTF–475, Revision 1, ''Control Rod Notch Testing Frequency and SRM Insert Control Rod Action.'' TSTF–475, Revision 1 modifies NUREG–1433 (BWR/4) and NUREG–1434 (BWR/6) STS. The changes: (1) revise TS testing frequency for surveillance requirement (SR) 3.1.3.2 in TS 3.1.3, ''Control Rod OPERABILITY," (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, Required Action E.2, ''Source Range Monitoring Instrumentation'' (NUREG–1434 only), and (3) revise Example 1.4–3 in Section 1.4 ''Frequency'' to clarify the applicability of the 1.25 surveillance test interval extension. The consequences of an accident after adopting TSTF–475, Revision 1 are no different than the consequences of an accident prior to adoption. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously analyzed. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

TSTF–475, Revision 1 will: (1) [revise the TS SR 3.1.3.2 frequency in TS 3.1.3, ‘‘Control Rod OPERABILITY’’, (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, ‘‘Source Range Monitoring Instrumentation,’’ and (3)] revise Example 1.4–3 in Section 1.4 ‘‘Frequency’’ to clarify the applicability of the 1.25 surveillance test interval extension. [The GE Nuclear Energy Report, ‘‘CRD Notching Surveillance Testing for Limerick Generating Station,’’ dated November 2006, concludes that extending the control rod notch test interval from weekly to monthly is not expected to impact the reliability of the scram system and that the analysis supports the decision to change the surveillance frequency.] Therefore, the proposed changes in TSTF–475, Revision 1 are acceptable and 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal

Nine Mile Point Nuclear Station, LLC, (NMPNS) Docket No. 50-220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: August 18, 2008

Description of amendment request: The proposed amendment would revise the NMP1 Technical Specification (TS) Section 3/4.1.1, "Control Rod System," to increase the Surveillance Requirement (SR) frequency associated with control rod exercising. The proposed change would revise the required SR frequency from once each week to once every 31 days. The proposed change is consistent with Nuclear Regulatory Commission (NRC)-approved Revision 1 to TS Task Force (TSTF) Change Traveler, TSTF-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," and NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 3.1. The availability of the TS improvement was announced in the Federal Register on November 13, 2007 (72 FR 63943) as part of the consolidated line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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:

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

The proposed change generically implements TSTF–475, Revision 1, ‘‘Control Rod Notch Testing Frequency and SRM Insert Control Rod Action.’’ TSTF–475, Revision 1 modifies NUREG–1433 (BWR/4) and NUREG–1434 (BWR/6) STS. The changes: (1) revise TS testing frequency for surveillance requirement (SR) 3.1.3.2 in TS 3.1.3, ‘‘Control Rod OPERABILITY,” (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, Required Action E.2, ‘‘Source Range Monitoring Instrumentation’’ (NUREG–1434 only), and (3) revise Example 1.4–3 in Section 1.4 ‘‘Frequency’’ to clarify the applicability of the 1.25 surveillance test interval extension. The consequences of an accident after adopting TSTF–475, Revision 1 are no different than the consequences of an accident prior to adoption. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously analyzed. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

TSTF–475, Revision 1 will: (1) [revise the TS SR 3.1.3.2 frequency in TS 3.1.3, ‘‘Control Rod OPERABILITY’’, (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, ‘‘Source Range Monitoring Instrumentation,’’

and (3)] revise Example 1.4–3 in Section 1.4 ‘‘Frequency’’ to clarify the applicability of the 1.25 surveillance test interval extension. [The GE Nuclear Energy Report, ‘‘CRD Notching Surveillance Testing for Limerick Generating Station,’’ dated November 2006, concludes that extending the control rod notch test interval from weekly to monthly is not expected to impact the reliability of the scram system and that the analysis supports the decision to change the surveillance frequency.] Therefore, the proposed changes in TSTF–475, Revision 1 are acceptable and 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal

Nuclear Management Company, LLC, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment request: July 11, 2008

Description of amendment request: The proposed amendments would establish Conditions, Required Actions, and Completion Times in the Prairie Island Nuclear Generating Plant, Units 1 and 2, Technical Specifications (TSs) for the condition where one steam supply to the turbine-driven auxiliary feedwater (AFW) pump is inoperable concurrent with an inoperable motor driven AFW train. The proposed amendments would also make changes to the TSs that establish

specific Actions for when the turbine-driven AFW train is inoperable either (a) due solely to one inoperable steam supply, or (b) due to reasons other than the one inoperable steam supply.

The NRC staff issued a notice of opportunity for comment in the Federal Register on March 19, 2007 (72 FR 12845), on possible amendments concerning the consolidated line item improvement process (CLIIP), including a model safety evaluation and a model no significant hazards consideration (NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the Federal Register on July 17, 2007 (72 FR 39089), as part of the CLIIP. In its application dated July 11, 2008, the licensee affirmed the applicability of the following determination.

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

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

   Response: No

   The Auxiliary/Emergency Feedwater (AFW/EFW) System is not an initiator of any design basis accident or event, and therefore the proposed changes do not increase the probability of any accident previously evaluated. The proposed changes to address the condition of one or two motor driven AFW/EFW trains inoperable and the turbine driven AFW/EFW train inoperable due to one steam supply inoperable do not change the response of the plant to any accidents.

   The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems, and components (SSCs) to perform their intended safety 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 any accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

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

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

Response: No

The proposed changes do not result in a change in the manner in which the AFW/EFW System provides plant protection. The AFW/EFW System will continue to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators. There are no design changes associated with the proposed changes. The changes to the Conditions and Required Actions 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 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?

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 impacted by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis.

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

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: Peter M. Glass, Assistant General Counsel, Xcel Energy Services, Inc., 414 Nicollet Mall, Minneapolis, MN 55401

NRC Branch Chief: Lois M. James

Tennessee Valley Authority, Docket No. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendments: September 4, 2008.

Brief description of amendments: The proposed amendment will delete the Technical specification (TS) requirements related to hydrogen recombiners and hydrogen monitors. Licensees were generally required to implement upgrades as described in NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," and Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the *Federal Register* on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated September 4, 2008.

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

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

The revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to 17 approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization or the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-

accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

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

NRC Section Chief: L. Raghavan

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

Date of amendment request: January 14, 2008

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) requirements related to control room envelope habitability in accordance with TS Task Force (TSTF) traveler TSTF-448-A, “Control Room Habitability,” Revision 3.

The NRC staff issued a "Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process" in the *Federal Register* on January 17, 2007 (72 FR 2022). The notice referenced a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request published in the *Federal Register* on October 17, 2006 (71 FR 61075). In its application dated January 14, 2008, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

> Criterion 1 - The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated
>
> The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that 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 significant reduction in a margin of safety.

The NRC staff has reviewed the analysis adopted by the licensee and, based upon this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendment involves NSHC.

Attorney for licensee: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, N.W., Washington, D.C. 20037

NRC Branch Chief: Michael T. Markley

PREVIOUSLY PUBLISHED NOTICES OF

CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the *Federal Register* on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: July 30, 2008

Description of amendment request: This amendment revises the Indian Point Nuclear Generating Unit No. 2 Technical Specification 3.8.1, Required Action A.4, to allow a one time extension to the completion time for the loss of one offsite power circuit from 72 hours to 144 hours. This change will ensure that there is enough time for the failed oil cooling pump on the station auxiliary transformer to be removed, and for the new oil cooling pump to be installed and tested.

Date of publication of individual notice in FEDERAL REGISTER: August 27, 2008

Expiration date of individual notice: October 27, 2008

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 51.22(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 email to pdr.resource@nrc.gov.

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

Date of amendment request: March 10, 2008, as supplemented by letters dated June 30, 2008, and September 29, 2008.

Description of amendment request: The amendment revised the Oyster Creek Technical Specifications (TSs) 3.3, "Reactor Coolant." Specifically, the amendment relocated the pressure and temperature limit curves to the licensee controlled document, "Pressure and Temperature Limits Report," (PTLR). Additionally, the amendment introduced supporting definitions and adds controls regarding the PTLR to Section 6.0, "Administrative Controls."

Date of issuance: September 30, 2008

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

Amendment No.: 269

Facility Operating License No. DPR-16: The amendment revised the License and Technical Specifications

Date of initial notice in FEDERAL REGISTER: June 17, 2008 (73 FR 34339). The supplemental letters provided additional information that clarified the application, did not expand

the scope of the application as originally noticed, and did not change the staff's initial proposed no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 2008.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket Nos. 50-336 and 50-423, Millstone Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendment: August 15, 2007, as supplemented on May 27, 2008, July 24, 2008, and September 3, 2008.

Brief description of amendment: The proposed amendment modified Technical Specification (TS) 3.3.3.1, "Radiation Monitoring," TS 3.4.6.1, "Reactor Coolant System Leakage Detection Systems," and Surveillance Requirements 4.4.6.1, "Reactor Coolant System Leakage Detection Systems." Specifically, the proposed amendment removed credit for the gaseous radiation monitor for Reactor Coolant System leakage detection. Improvements in nuclear fuel reliability over time have resulted in the reduction of effectiveness of the monitors in detecting very small leaks and very small changes in the leak rate. The proposed change also addressed the condition when the remaining monitoring systems are all inoperable.

Date of issuance: September 30, 2008

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

Amendment Nos.: 306 and 244

Renewed Facility Operating License Nos. DPR-65 and NPF-49: Amendment revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: June 17, 2008, (73 FR 34341)

The supplements dated May 27, 2008, July 24, 2008, and September 3, 2008, clarified the application, did not expand the scope of the application as originally noticed, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 2008.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50-423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: May 8, 2008, as supplemented by letter dated August 14, 2008

Brief description of amendment: This amendment request contains sensitive unclassified non-safeguards information. The changes allow for interim alternate steam generator tube repair criterion, as specified in the Millstone Power Station, Unit 3 (MPS3) technical specifications. The interim alternate repair criterion is for the upcoming refueling outage and the subsequent operating cycle. The amendment also adds three reporting criteria to the MPS3 technical specifications for steam generator tube inspections.

Date of issuance: September 30, 2008

Effective date: As of the date of issuance and shall be implemented prior to Mode 5 startup

Amendment No.: 245

Renewed Facility Operating License No. NPF-49: Amendment revised the License and Technical Specifications.

Date of initial notice in FEDERAL REGISTER: July 8, 2008 (73 FR 39054)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 2008.

No significant hazards consideration comments received: No.

Duke Energy Carolinas, LLC, et al., Docket Nos. 50-413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: December 20, 2007

Brief description of amendment: The amendment reflects the direct transfer of the undivided ownership interest of the Saluda River Electric Cooperation, Inc, in Catawba Nuclear Station, Unit 1, to Duke Energy Carolinas, LLC, a current owner and operator, and the North Carolina Electric Membership Corporation, a current owner.

Date of issuance: September 30, 2008

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

Amendment No.: 245

Facility Operating License Nos. NPF-35: Amendment revised the license.

Date of initial notice in *FEDERAL REGISTER*: July 21, 2008 (73 FR 42375)

The supplement dated May 29, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 25, 2008.

No significant hazards consideration comments received: No

Energy Northwest, Docket No. 50-397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: July 26, 2007, as superseded by application dated August 8, 2007, and as supplemented by letters dated November 19, 2007, and June 5 and July 21, 2008.

Brief description of amendment: The amendment revises the requirements of Technical Specification (TS) 3.3.5.2, "Reactor Core Isolation Cooling (RCIC) System Instrumentation," and TS 3.5.2, "ECCS [Emergency Core Cooling System]-Shutdown," to increase the Condensate Storage Tank level.

Date of issuance: September 30, 2008

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

Amendment No.: 210

Facility Operating License No. NPF-21: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *FEDERAL REGISTER*: August 28, 2007 (72 FR 49572)

The supplements dated November 19, 2007, and June 5 and July 21, 2008, 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 September 30, 2008.

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: April 22, 2008, as supplemented by letters date July 2, July 22, and September 24, 2008.

Brief description of amendment: The amendment modified Technical Specification (TS) 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability Section 3.4.9, "RCS [Reactor Coolant System] Pressure and Temperature (P-T) Limits," and Section 5.0, "Administrative Controls," to delete reference to the pressure and temperature curves, and include reference to the Pressure and Temperature Limits Report (PTLR). This change adopted the methodology of SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," for preparation of the pressure and temperature curves, and incorporated the guidance of TSTF-419-A, "Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS PTLR."

Date of issuance: October 3, 2008

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

Amendment No.: 292

Facility Operating License No. DPR-59: The amendment revised the License and the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: July 1, 2008, (73 FR 37503)

The supplemental submissions dated July 2, July 22, and September 24, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2008.

No significant hazards consideration comments received: No

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

Date of application for amendment: June 17, 2008.

Brief description of amendment: The amendments revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.9, "Steam Generator (SG) Tube Inspection Report." For TS 5.5.9, the amendments incorporate a one-cycle interim alternate repair criteria in the provisions for SG tube repair criteria during Byron, Unit No. 2, refueling outage 14 and the subsequent operating cycle. For TS 5.6.9, the amendments revise the current reporting requirements. These changes only affect Byron, Unit No. 2; however, this action is docketed for both Byron units because the TS are common to both units.

Date of issuance: October 1, 2008

Effective date: As of the date of issuance and shall be implemented prior to the return to service from Byron, Unit No. 2, fall 2008 Refueling Outage 14.

Amendment Nos.: Unit 1 -158; Unit 2 -158

Facility Operating License Nos. NPF-37 and NPF-66: The amendment revised the TSs and License.

Date of initial notice in FEDERAL REGISTER: August 5, 2008 (73 FR 45485).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 2008.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: July 16, 2007, as supplemented May 20 and August 26, 2008

Brief description of amendments: Amendments modified the technical specification requirements related to control room envelope habitability in accordance with Technical Specification Task Force (TSTF) Traveler TSTF-448, Revision 3, "Control Room Habitability."

Date of Issuance: September 30, 2008.

Effective Date: Unit 1 - Amendment is effective as of the date of its issuance and shall be implemented following implementation of the Amendment No. 152, regarding Alternative Source Term and with the completion of the installation and testing of the plant modifications described in the licensee's application, including letters dated July 16, 2007, February 14, March 18, April 14, June 2, July 11, and August 13, 2008. Unit 2 - This license amendment is effective as of the date of its issuance and shall be implemented following implementation of License Amendment No. 152.

Amendment Nos.: 205 and 153.

Renewed Facility Operating License Nos. DPR-67 and NPF-16: Amendments revised the Technical Specifications.

Date of initial notice in *Federal Register*: August 28, 2007 (72 FR 49578). The supplements dated May 20 and August 26, 2008, 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 September 30, 2008.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 16, 2007, as supplemented by letters dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008.

Brief description of amendment: The amendment modifies the facility's operating licensing bases to adopt the alternative source term as allowed in 10 CFR 50.67, and as described in Regulatory Guide 1.183. The licensee revised the plant licensing basis through reanalysis of the radiological consequences of the following Updated Final Safety Analysis Report Chapter 15 accidents: Loss-of-Coolant Accident, Fuel-Handling Accident, Main Steam Line Break, Steam Generator Tube Rupture, Reactor Coolant Pump Shaft Seizure, Control Element Assembly Ejection, Letdown Line Break, and Feedwater Line Break.

Date of issuance: September 29, 2008.

Effective date: Effective as of the date of issuance and shall be implemented within 180 days.

Amendment No.: 152.

Renewed Facility Operating License No. NPF-16: The amendment revises the Technical Specifications and the Renewed Facility Operating License.

Date of initial notice in *Federal Register*: June 12, 2008 (73 FR 33460). The supplements dated February 14, March 18, April 14, June 2, July 11, and August 13, 2008, 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.

Public comments received as to proposed no significant hazards consideration (NSHC): No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 29, 2008.

Attorney for licensee: M. S. Ross, Managing Attorney, Florida Power and Light Company, P.O. Box 14000, Juno Beach, Florida 33408-0420.

NRC Branch Chief: Thomas H. Boyce.

Nine Mile Point Nuclear Station, LLC, Docket Nos. 50-220 and 50-410, Nine Mile Point Nuclear Station, Unit Nos. 1 and 2 (NMP1 and NMP2), Oswego County, New York

Date of application for amendment: December 20, 2007

Brief description of amendments: The amendments revise NMP1 Technical Specification (TS) Section 6.3, "Unit Staff Qualifications," and NMP2 TS Section 5.3, "Unit Staff Qualifications," to update requirements that have been superseded due to the accreditation of the NMPNS licensed operator training program and due to promulgation of the revised Title 10 of the *Code of Federal Regulations* (10 CFR), Part 55, "Operators' Licenses," which became effective on May 26, 1987 (52 FR 9453). Additionally, the amendment for NMP1 revises the TSs by eliminating the qualification requirement exceptions listed for the position of Manager Operations which were previously approved by the NRC staff. The position of Manager Operations would meet the minimum qualification requirements as required in American National Standard Institute Standard NI8.1-1971, "American National Standard for Selection and Training of Nuclear Power Plant Personnel."

Date of issuance: September 29, 2008

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

Amendment Nos.: 198 and 127

Renewed Facility Operating License No. DPR-63 and NPF-069: Amendments revise the License and TSs.

Date of initial notice in FEDERAL REGISTER: January 28, 2008 (73 FR 5225)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 29, 2008.

No significant hazards consideration comments received: No

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: October 3, 2007

Brief description of amendments: The amendments revised a footnote in Technical Specifications Table 3.3.2.1-1, "Control Rod Block Instrumentation," such that a new banked position withdrawal sequence shutdown sequence could be utilized. Associated changes are made to the TS Bases. This operating license improvement was made available by the NRC staff on May 23, 2007, as part of the consolidated line item improvement process.

Date of issuance: October 1, 2008.

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

Amendment Nos.: Unit 1 – 258, Unit 2 - 202

Renewed Facility Operating License Nos. DPR-57 and NPF-5: Amendments revised the licenses and the technical specifications.

Date of initial notice in *FEDERAL REGISTER*: November 6, 2007 (72 FR 62691)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 2008.

No significant hazards consideration comments received: No

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: October 5, 2007.

Brief description of amendments: The amendments revise the TSs completion times (CTs) for TS Limiting Condition of Operation (LCO) 3.8.1, Conditions B and C, by specifying when maintenance restrictions need to be met and by adding a 72-hour CT for the swing DG 1B.

Date of issuance: October 2, 2008.

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

Amendment Nos.: Unit 1 - 259, Unit 2 – 203.

Renewed Facility Operating License Nos. DPR-57 and NPF-5: Amendments revised the licenses and the technical specifications.

Date of initial notice in *Federal Register*: November 6, 2007, (72 FR 62691)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 2008.

No significant hazards consideration comments received: No

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

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

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 application for amendments: June 12, 2008

Brief description of amendments: The amendments revised the Technical Specifications requirement for the Plant Manager or the Operations Manager regarding the holding of a Senior Reactor Operator license.

Date of issuance: October 7, 2008

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

Amendment Nos.: Farley Unit 1 - 179; Unit 2 - 171; Hatch Unit 1 - 260; Unit 2 - 204; Vogtle Unit 1 - 153; Unit 2 – 134

Facility Operating License Nos. NPF-2 and NPF-8; DPR-57 and NPF-5; NPF-68 and NPF-81: Amendments revised the licenses and the technical specifications.

Date of initial notice in *FEDERAL REGISTER*: July 1, 2008, 73 FR 37505

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 7, 2008.

No significant hazards consideration comments received: No

Tennessee Valley Authority, Docket No. 50-327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendment: April 14, 2008.

Brief description of amendment: The amendment revises the list of topical reports referenced in Technical Specification Section 6.9.1.14.a for use in preparing the core operating limits report by adding EMF-2103P-A, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." The change will be utilized in core loading designs for Unit 1 fuel-load configurations in future operating cycles.

Date of issuance: September 24, 2008.

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

Amendment No.: 320.

Facility Operating License No. DPR-77: Amendment revises the technical specifications.

Date of initial notice in *Federal Register*: June 10, 2008 (73 FR 32746). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 2008.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 10th day of October 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joseph Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation