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 July 22, 2005, to August 4, 2005. The last biweekly notice was published on August 2, 2005 (70 FR 44400).

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

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

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

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

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

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

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

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

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

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

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

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

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

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: June 20, 2005.

Description of amendments request: The proposed change would revise the Technical Specification Surveillance Requirement 3.6.1.6.2 of 3.6.1.6, ''Suppression Chamber-to-Drywell Vacuum Breakers'' for the frequency of functionally testing the suppression chamber-to-drywell vacuum breakers.

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

The proposed change revises Surveillance Requirement [SR] 3.6.1.6.2 to require performance of functional testing of each suppression chamber-to-drywell vacuum breaker every 92 days, within 12 hours after any discharge of steam to the suppression chamber from the safety/relief valves, and within 12 hours following an operation that causes any of the vacuum breakers to open.

The proposed change does not involve physical changes to any plant structure, system, or component. The suppression chamber-to-drywell vacuum breakers only provide an accident mitigation function. As such, the probability of occurrence for a previously analyzed accident is not impacted by the change to the surveillance frequency for these components. The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability of successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. No physical change to suppression chamber-to-drywell vacuum breakers is being made as a result of the proposed change, nor does the change alter the manner in which the vacuum breakers operate. As a result, no new failure modes of the suppression chamber-to-drywell vacuum breakers are being introduced. The proposed quarterly surveillance frequency for the suppression chamber-to-drywell vacuum breakers is consistent with the American Society of Mechanical Engineers (ASME) Code frequency for testing these valves, will avoid unnecessary cycling and wear of the vacuum breakers, and will improve the reliability of the vacuum breakers. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

Therefore, the proposed change to the surveillance frequency for the suppression chamber-to-drywell vacuum breakers does not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Does not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed change to the surveillance frequency for the suppression chamber-to-drywell vacuum breakers does not involve any physical alteration of plant systems, structures, or components. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. Therefore, the proposed change to the surveillance frequency for the suppression chamber-to-drywell vacuum breakers does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does not involve a significant reduction in the margin of safety.

The proposed change revises SR 3.6.1.6.2 to require performance of functional testing of each vacuum breaker every 92 days, within 12 hours after any discharge of the steam to the suppression chamber from the safety/relief valves, and within 12 hours following an operation that causes any of the vacuum breakers to open. The operability and functional characteristics of the suppression chamber-to-drywell vacuum breakers remains unchanged. The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through the margins contained within the safety analyses. The proposed change to the surveillance frequency for the suppression chamber-to-drywell vacuum breakers does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. As previously noted, the proposed quarterly surveillance frequency for the suppression chamber-to-drywell vacuum breakers is consistent with the ASME Code for frequency for testing these vacuum breakers, will avoid unnecessary cycling and wear of the vacuum breakers, and will improve the reliability of the vacuum breakers. The proposed change does not impact any safety analysis assumptions or results. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: June 29, 2005.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) to revise Surveillance Requirements (SR) 3.6.1.3.11 and 3.6.1.3.12 in TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs).'' Specifically, the proposed amendment would revise the combined secondary containment bypass leakage rate limit for all bypass leakage paths in SR 3.6.1.3.11 from 0.05 to 0.10 L_a and the combined main steam isolation valve (MSIV) leakage rate limit for all four main steam lines in SR 3.6.1.3.12 from 150 to 250 standard cubic feet per hour (scfh).

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

The increase in the allowed secondary containment bypass leakage limit in SR 3.6.1.3.11 and the increase in the total Main Steam Isolation Valve (MSIV) leakage rate limit have been evaluated in a revision to the analysis of the Loss of Coolant Accident (LOCA). Based on the results of the analysis, it has been demonstrated that, with the requested change, the dose consequences of this limiting Design Basis Accident (DBA) are within the regulatory guidance provided by the NRC [Nuclear Regulatory Commission] for use with the AST [alternative source term]. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183, ''Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors,'' and Standard Review Plan (SRP) Section 15.0.1. The proposed change also updates the design basis value for the Control Room Envelope (CRE) unfiltered inleakage based on actual test results. This is acceptable because the assumed value in the analysis is more than three times the worst case test value. The proposed change does not affect the normal design or operation of the facility before the accident; rather, it affects leakage limit assumptions that constitute inputs to the evaluation of the consequences. The radiological consequences of the analyzed LOCA have been evaluated using the plant licensing basis for this accident. The results conclude that the control room and offsite doses remain within applicable regulatory limits. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The change in leakage limits does not affect the design, functional performance or normal operation of the facility. Similarly, it does not affect the design or operation of any component in the facility such that new equipment failure modes are created. As such the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed license amendment involves changes in leakage rate limits for the secondary containment bypass leakage and MSIV leakage. The revised leakage rate limits are used in the LOCA radiological analysis in conjunction with the revised CRE unfiltered inleakage limit. The analysis has been performed using conservative methodologies. Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed LOCA event has been carefully selected and margin has been retained to ensure that the analysis adequately bounds postulated event scenario. The dose consequences of this limiting event are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183 and SRP Section 15.0.1. The margin of safety is that provided by meeting the applicable regulatory limits. The effect of the revision to the Technical Specification requirements has been analyzed and doses resulting from the pertinent design basis accident have been found to remain within the regulatory limits. The change continues to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

Entergy Nuclear Operations, Inc., Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3, Westchester County, New York

Date of amendment request: June 8, 2005.

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

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated June 8, 2005.

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

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in Regulatory Guide 1.177. A bounding risk assessment was performed to justify the proposed TS changes. The proposed LCO 3.0.8 defines limitations on the use of the provision and includes a requirement for the licensee to assess and manage the risk associated with operation with an inoperable snubber. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: May 31, 2005.

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

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated May 31, 2005.

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

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

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

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

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

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

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

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

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

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: May 24, 2005.

Description of amendment request: The proposed amendment would delete the Technical Specification (TS) temperature limit for the safety relief valve (SRV) discharge pipe and the requirements for NRC approval of the associated engineering evaluation.

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. This proposed change deletes an administrative requirement for NRC approval of an engineering evaluation to resolve a non-conforming and degraded condition that is required by NRC Generic Letter 91–18 (GL), Rev. 1, ''Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions''. The SRVs will be maintained operable, inspected, and tested to perform their safety function as required by the current Specifications and any SRV non-conforming or degraded condition will be addressed in accordance with GL 91–18. The proposed change also deletes a Note regarding installed two-stage Target Rock SRVs. The deletion of an administrative requirement and the Note does not change the plant response to the design basis accident and does not increase the probability of inadvertent SRV operation. Therefore, the proposed change does not significantly increase the probability or consequences of any previously evaluated accidents.

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

Response: No. The safety function of the SRVs is to provide over-pressure protection of the primary coolant pressure boundary and also for the automatic functions to rapidly depressurize the primary system to a pressure at which low-pressure cooling systems can provide makeup. The proposed change deletes an administrative requirement and a Note related to installed two-stage Target Rock SRVs, and does not introduce any new modes of equipment operation or failure. 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 ability of the SRVs to perform their safety function is maintained during operation and will continue to be tested as required in accordance with TS 3/4.13, Inservice Code Testing. The proposed change deletes an administrative requirement that is adequately addressed by following GL 91–18, Rev. 1. Deletion of an administrative requirement does not reduce the margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: Darrell Roberts.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: May 24, 2005.

Description of amendment request: The proposed amendment would delete the main steam isolation valve (MSIV) twice per week partial stroke testing surveillance specified in Technical specification (TS) 4.7.A.2.b.1.c.

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. This proposed change deletes the requirement to exercise the MSIV's twice per week at power. The MSIVs will continue to be full stroke tested by the Inservice Testing Program. The MSIVs will continue to be able to perform their accident mitigation function. The plant response to the design basis accident will not change and the probability of inadvertent MSIV closure will not be increased. Therefore, the proposed change does not significantly increase the probability or consequences of any previously evaluated accidents.

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

Response: No. The safety function of the MSIVs is to isolate the main steam lines in case of design basis accidents to limit the loss of reactor coolant and/or limit the release of radioactive materials. The proposed change does not introduce any new modes of equipment operation or failure. 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 ability of the MSIVs to perform their safety function is tested during the MSIV full stroke fast closure test in accordance with TS 3.13, Inservice Testing Program. The proposed change deletes a high-risk surveillance. Deletion of the high-risk surveillance does not reduce the margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: Darrell Roberts.

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: March 7, 2005.

Description of amendment request: The proposed amendment request will add two NRC approved topical report references to the list of analytical methods in Technical Specification 5.6.5, ''Core Operating Limits Report (COLR),'' that can be used to determine core operating limits. The proposed changes are:

1. Add a NRC previously approved Siemans Power Corporation (SPC) topical report reference for determination of fuel assembly critical power for previously loaded Global Nuclear Fuel (GNNF) GE14 fuel which will be co-resident with reloaded Framatome ANP ATRIUM–10 fuel.

2. Add a NRC previously approved Framatome Advanced Nuclear Power, Inc. (FRA–ANP) topical report reference for an uprated methodology for evaluation of loss coolant accident (LOCA) conditions.

The proposed changes are the result of a redesign to untilize Framatome ANP ATRIUM–10 fuel during the Unit 1 Refueling Outage 11 currently scheduled for February 2006.

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

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

Response: No.

The proposed changes will add two additional NRC approved topical report references to the list of administratively controlled analytical methods in Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR),'' that can be used to determine core operating limits. TS 5.6.5 lists NRC approved analytical methods used at LaSalle County Station (LSCS) to determine core operating limits.

LSCS Unit 1 is scheduled to reload Framatome ANP ATRIUM–10 fuel during the Unit 1 Refueling Outage 11currently scheduled for February 2006. The proposed changes to TS Section 5.6.5 will add FRA-ANP methodologies to determine overall core operating limits for future core configurations. This change will require the listing of additional analytical methods for evaluating LOCA conditions and determining the critical power performance of the GE14 fuel. Thus, the proposed changes will allow LSCS to use the most recent FRA-ANP LOCA methodology for evaluation of ATRIUM–10 fuel and SPC critical power correlations to determine the critical power for the GE14 fuel.

The addition of approved methods to TS Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The methods have been reviewed to ensure that the output accurately models predicted core behavior, have no effect on the type or amount of radiation released, and have no effect on predicted offsite doses in the event of an accident. Additionally the methods do not change any key core parameters that influence any accident consequences. Thus, the proposed changes do not have any effect on the probability of an accident previously evaluated.

The methodology conservatively establishes acceptable core operating limits such that the consequences of previously analyzed events are not significantly increased.

The proposed changes in the administratively controlled analytical methods do not affect the ability of LSCS to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

Response: No.

The proposed changes involve TS 5.6.5 do not affect the performance of any LSCS structure, system, or component credited with mitigating any accident previously evaluated. The insertion of fuel, which has been analyzed with NRC approved methodologies, will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed changes do not introduce any new modes of system operation or failure mechanism.

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

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

Response: No.

The proposed changes will add two additional references to the list of administratively controlled analytical methods in TS 5.6.5 that can be used to determine core operating limits. The proposed changes do not modify the safety limits or setpoints at which protective actions are initiated and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. Therefore, LSCS has determined that the proposed changes provide an equivalent level of protection as that currently provided.

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

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

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

NRC Section Chief: Gene Y. Suh.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: July 5, 2005.

Description of amendment request: The proposed amendment would modify the existing Technical Specification (TS) 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' Surveillance Requirement (SR) 3.3.1.3.5. Specifically, the thermal power level at which the OPRMs are ''not bypassed'' (enabled to perform their design function) will be changed from > 28.6 percent rated thermal power to ≥ 23.8 percent rated thermal power.

Plant-specific stability calculations are now required as part of the resolution to several generic issues associated with OPRM operability. One of the outcomes from this resolution was a change in the OPRM enabled region of the power to flow map. The thermal power level for enabling the OPRMs for Cycle 10 became > 27.2 percent rated thermal power. Since the current TS SR requirement is > 28.6 percent, the new TS SR thermal power level value is considered a non-conservative TS. The Perry Nuclear Power Plant (PNPP) is currently requiring the OPRMs to be enabled at ≥ 23.8 percent thermal power level through administrative controls. These controls will remain in place until such time that this license amendment is approved (reference NRC Administrative Letter 98–10, ''Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety,'' dated December 12, 1998).

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

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

The proposed change involves the use of a revised thermal power level to establish the OPRM enabled region. The OPRM enabled region is that area on the power to flow map where the OPRM System is activated to detect and suppress potential instability events. If reactor operations result in entrance into this region and a core instability is detected, the OPRM System will automatically initiate a reactor scram. The revised enabled region provides assurance that the requirements of 10CFR50, Appendix A, General Design Criteria 10 and 12 remain satisfied for current and future core designs. Though the initiation of instability events are dependent upon thermal power levels and core flows, the revision to the enabled region thermal power level value does not increase the possibility of such an event. Once the OPRMs are enabled, the OPRM System would still mitigate an instability event, if detected. The revised enabled region does not impact any OPRM detection or mitigation actions for instability events.

The OPRMs are designed to detect and suppress potential instability events. As such, the OPRMs are not credited to provide any type of detection or mitigation actions for transients or accidents described within the PNPP Updated Final Safety Analysis Report (USAR) other than instability events. Hence, revising the OPRMs enabled region will not impact the transients or accidents described within the PNPP Updated Safety Analysis Report (USAR) other than instability events.

Since the OPRMs will be enabled at a thermal power lower than analytically required, the potential for additional scrams exists. However, since the possibility of an instability event occurring in the range between the revised thermal power level and the analytical value is remote, the probability of an additional scram from occurring is not significantly increased.

Therefore, since no significant changes are being made to the plant or its design, the probability or the consequences of an accident have not increased over those previously evaluated.

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 involves the use of a revised thermal power level to establish the OPRM enabled region. The use of a revised thermal power level to establish the OPRM enabled region does not involve a physical modification to any plant system or component, including the fuel. The revised enabled region provides assurance that the requirements of 10CFR50, Appendix A, General Design Criteria 10 and 12 remain satisfied for current and future core designs. Though the initiation of instability events are dependent upon thermal power levels and core flows, the revision to the enabled region thermal power level value does not increase the possibility of such an event, or introduce any new or different events. Once the OPRMs are enabled, the OPRM System detects and mitigates an instability event if detected. The revised enabled region does not impact any mitigation actions.

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

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

The proposed change involves the use of a revised thermal power level to establish the OPRM enabled region. Once the OPRMs are enabled, the OPRM System mitigates an instability event if detected. The revised enabled region does not impact any mitigation actions. The use of a revised thermal power level to establish the OPRM enabled region does not involve a physical modification to any plant system or component, including the fuel. The revised enabled region provides assurance that the requirements of 10CFR50, Appendix A, General Design Criteria 10 and 12 remain satisfied for current and future core designs. The revised enabled region restores the margin of protection provided by the OPRMs, which had been reduced as fuel and core designs have evolved since 1994. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Gene Y. Suh.

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 amendment request: May 25, 2005.

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

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated May 25, 2005.

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

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

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

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

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

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

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

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and

Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Evangelos C. Marinos.

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

Date of amendment request: July 5, 2005.

Description of amendment request: The proposed changes to the Technical Specifications (TS) would add a reference in TS 5.65.b, ''Core Operating Limits Report (COLR),'' to permit the use of an alternate methodology, VIPRE–D/BWU code/correlation (Virginia Electric and Power Company version of the Electric Power Research Institute (EPRI) computer code VIPRE [Versatile Internals and Components Program for Reactors—EPRI] with the BWU Critical Heat Flux (CHF) correlations), to perform thermal-hydraulic analysis to predict CHF and Departure from Nucleate Boiling Ratio (DNBR) for the AREVA Advanced Mark-BW (AMBW) fuel in the North Anna cores.

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

1. The probability of occurrence or the consequences of an accident previously evaluated are not significantly increased.

Neither the code/CHF correlation pair nor the Statistical DNBR Evaluation Methodology make any contribution to the potential accident initiators and thus cannot increase the probability of any accident. Further, since both the deterministic and statistical DNBR limits meet the required design basis of avoiding DNB with 95% probability at a 95% confidence level, the use of the new code/correlation and Statistical DNBR Evaluation Methodology do not increase the potential consequences of any accident. Finally the addition of a full core DNB design limit provides increased assurance that the consequences of a postulated accident which included radioactive release would be minimized because the overall number of rods in DNB would not exceed the 0.1% level. All the pertinent evaluations to be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable have been performed with VIPRE–D/BWU and found to be acceptable. The use of a different code/correlation pair will not increase the probability of an accident because plant systems will not be operated in a different manner, and system interfaces will not change. The use of the VIPRE–D/BWU code/correlation pair will not result in a measurable impact on normal operating plant releases, and will not increase the predicted radiological consequences of accidents postulated in the UFSAR [Updated Final Safety Analysis Report]. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

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

The use of VIPRE–D/BWU and its applicable fuel design limits for DNBR does not impact any of the applicable design criteria and all pertinent licensing basis criteria will continue to be met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Setpoint safety analysis evaluations have demonstrated that the use of VIPRE–D/BWU is acceptable. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of VIPRE–D/BWU code/correlation or the Statistical DNBR Evaluation Methodology does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced. North Anna Technical Specification 2.1 specifies that any DNBR limit Established by any used code/correlation must provide at least 95% non-DNB probability at a 95% confidence level. The use of VIPRE–D/BWU with the SDLs [Statistical Design Limits] listed in this package provides that protection, just as LYNXT/BWU [LYNXT thermal-hydraulic computer code with the AREVA BWU CHF correlations] and applicable SDLs did. The required DNBR margin of safety for the North Anna Nuclear units, which in this case is the margin between the 95/95 DNBR limit and clad failure, is therefore not reduced. Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

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

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

NRC Section Chief: Evangelos C. Marinos.

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

Date of amendment request: July 14, 2005.

Description of amendment request: The proposed changes to the Technical Specifications (TS) would correct two errors in the units of measure used to determine the Overtemperature ΔT Function Allowable Value.

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

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. The proposed changes correct errors in the unit designations used in the $f_1(\Delta I)$ equation. The actual numerical values of $f_1(\Delta I)$ calculated by the equation remain the same, only the units applied to the value are changed. The Overtemperature ΔT function allowable values are utilized by the Reactor Trip System (RTS) instrumentation to prevent reactor operation in conditions outside the range considered for accident analyses. The proposed changes will not alter the allowable values used by the RTS instrumentation. The Overtemperature ΔT allowable value is not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. As the Overtemperature ΔT allowable value is not changed, the probability or consequences of an accident previously evaluated is not significantly increased.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident already evaluated in the UFSAR. The proposed changes correct errors in the unit designations used in the $f_1(\Delta I)$ equation. Changes do not introduce a new mode of plant operation and do not involve any physical modifications to the plant. The changes will not introduce new accident initiators. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety. The proposed changes correct errors in the unit designations used in the $f_1(\Delta I)$ equation. This will eliminate the possibility of an error resulting from incorrect interpretation of the equation and potential subsequent errors in the application of the equation. The allowable value of the Overtemperature ΔT function is unaffected. Therefore, the proposed changes will not significantly reduce the margin of safety as defined in the Technical Specifications.

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

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

NRC Section Chief: Evangelos C. Marinos.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: December 14, 2004.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.3.G, ''Scram Discharge Volume,'' for the condition of having one or more SDV vent or drain lines with inoperable valves.

Date of issuance: July 29, 2005.

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

Amendment No.: 216.

Facility Operating License No. DPR–35: The amendment revised the TSs.

*Date of initial notice in **Federal Register:*** May 24, 2005 (70 FR 29792).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 8, 2004.

Brief description of amendments: These amendments relocated several Technical Specifications (TSs) from Section 6, ''Administrative Controls,'' requirements to the Quality Assurance Topical Report. Specifically, the amendments relocated (1) the Plant Operations Review Committee and Nuclear Review Board requirements, (2) the program/procedure review and approval requirements, and (3) the record-retention requirements.

Date of issuance: July 25, 2005.

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

Amendment Nos.: 176 and 138.

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

*Date of initial notice in **Federal Register***: June 22, 2004 (69 FR 34701).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 22, 2005.

Brief description of amendments: The amendments revise Technical Specifications by eliminating the requirements to provide the NRC monthly operating reports and annual occupational radiation exposure reports.

Date of issuance: July 28, 2005.

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

Amendment Nos.: 266 and 148.

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

*Date of initial notice in **Federal Register***: May 10, 2005 (70 FR 24651).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 29, 2004.

Brief description of amendment: The amendment deleted the requirements from the technical specifications to maintain a hydrogen dilution system, a hydrogen purge system, and hydrogen monitors.

Date of issuance: August 1, 2005.

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

Amendment No.: 265.

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

*Date of initial notice in **Federal Register***: February 15, 2005 (70 FR 7764).

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

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: October 15, 2004.

Brief description of amendment: The amendment revises surveillance requirements related to the reactor coolant pump flywheel inspections to extend the allowable inspection interval to 20 years.

Date of issuance: July 27, 2005.

Effective date: July 27, 2005.

Amendment No.: 218.

Facility Operating License No. DPR–72: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9992).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 11, 2004.

Brief description of amendments: The amendments revise Technical Specification (TS) Surveillance Requirement 3.1.7.7 acceptance criteria from 1224 psig to 1395 psig in TS 3.1.7, ''Standby Liquid Control System.''

Date of issuance: July 25, 2005.

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

Amendment Nos.: 221, 198.

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

Date of initial notice in ***Federal Register:*** July 6, 2004 (69 FR 40678).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 8, 2004.

Brief description of amendments: The amendments revised Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' for the condition of having one or more SDV vent or drain lines with one or both valves inoperable.

Date of issuance: July 26, 2005.

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

Amendment Nos.: 222 and 199.

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

Date of initial notice in ***Federal Register:*** December 7, 2004 (69 FR 70721).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 26, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 8, 2004.

Brief description of amendments: The amendments revised SSES 1 and 2 Technical Specification (TS) Surveillance Requirement 3.6.1.3.6 of TS 3.6.1.3, ''Primary Containment Isolation Valves,'' to reduce the frequency of performing leakage rate testing for each primary containment purge valve with resilient seals from 184 days to 24 months.

Date of issuance: August 4, 2005.

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

Amendment Nos.: 223 and 200.

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

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9995).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 4, 2005.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259 Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendment: August 2, 2004 (TS–435).

Brief description of amendment: The amendment modifies the Technical Specification (TS) 3.6.3.1 required action to provide 7 days of continued operation with two Containment Atmosphere Dilution subsystems inoperable.

Date of issuance: July 18, 2005.

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

Amendment No.: 255.

Facility Operating License Nos. DPR–33: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 9, 2004 (69 FR 64991).

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

No significant hazards consideration comments received: No.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of amendment request: November 24, 2003, and supplemented by letters dated December 10, 2003, December 16, 2003, January 19, 2004, January 21, 2004, February 10, 2004, March 4, 2004, April 27, 2004, August 3, 2004, September 2, 2004, September 2, 2004, September 30, 2004, November 19, 2004, December 10, 2004, and April 7, 2005. Supplemental letters provided additional clarifying information that 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.

Description of amendment request: The amendment revises the license to incorporate a new license condition addressing the license termination plan (LTP). This amendment documents the approval of the LTP, documents the criteria for making changes to the LTP which will and will not require pre-approval by the NRC, and documents the conditions imposed with the approval of the LTP.

Date of issuance: July 28, 2005.

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

Amendment No.: 158.

Facility Operating License No. DPR–3: Amendment revises the license.

Date of initial notice in ***Federal Register:*** February 18, 2003 (68 FR 7823).

The Commission's related evaluation of the amendment, state consultation, and final NSHC determination are contained in a safety evaluation dated July 28, 2005.

No significant hazards consideration comments received: No.

NRC Section Chief: Claudia Craig.

Dated at Rockville, Maryland, this 8th day of August, 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. E5–4403 Filed 8–15–05; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

January 2005 Pay Adjustments

AGENCY: Office of Personnel Management.

ACTION: Notice.