NUCLEAR REGULATORY COMMISSION

[Docket No. 50–368]

Entergy Operations, Inc., Arkansas Nuclear One, Unit 2; Notice of Availability of the Final Supplement 19 to the Generic Environmental Impact Statement for the License Renewal of Arkansas Nuclear One, Unit 2

Notice is hereby given that the U.S. Nuclear Regulatory Commission (Commission) has published a final plant-specific supplement to the Generic Environmental Impact Statement (GEIS), NUREG–1437, regarding the renewal of operating license NPF–6 for an additional 20 years of operation at Arkansas Nuclear One, Unit 2 (ANO–2). ANO–2 is located in Pope County, Arkansas, approximately 6 miles west-northwest of Russellville, Arkansas. Possible alternatives to the proposed action (license renewal) include no action and reasonable alternative energy sources.

In Section 9.3 of the final Supplement 19 to the GEIS, the staff concludes that based on: (1) The analysis and findings in the GEIS; (2) the environmental report submitted by Entergy; (3) consultation with Federal, State, and local agencies; (4) the staff's own independent review; and (5) the staff's consideration of public comments received during the environmental review, the staff recommends that the Commission determine that the adverse environmental impacts of license renewal for ANO–2, are not so great that preserving the option of license renewal for energy-planning decisionmakers would be unreasonable.

The final Supplement 19 to the GEIS is available for public inspection in the NRC Public Document Room (PDR) located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland, or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room). Persons who do not have access to ADAMS, or who encounter problems in accessing the documents located in ADAMS, should contact the PDR reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.* In addition, the Ross Pendergraft Library at Arkansas Tech University, 305 West Q Street, Russellville, Arkansas 72801, has agreed to make the final plant-specific supplement to the GEIS available for public inspection.

FOR FURTHER INFORMATION CONTACT: Mr. Thomas Kenyon, License Renewal and Environmental Impacts Program, Division of Regulatory Improvement Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Mr. Kenyon may be contacted at 301–415–1120 or *TJK@nrc.gov.*

Dated in Rockville, Maryland, this 7th day of April, 2005.

For the Nuclear Regulatory Commission.

Pao-Tsin Kuo,

Program Director, License Renewal and Environmental Impacts Program, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 1, 2005, through April 14, 2005. The last biweekly notice was published on April 12, 2005 (70 FR 19110).

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

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

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

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

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

Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

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

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

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

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

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: February 25, 2005.

Description of amendment request: The proposed change would delete Section 2.G of the Clinton's Facility Operating License (FOL), NPF–62, which requires AmerGen Energy Company, LLC, to report violations of the requirements contained in Section 2.C of this license. The proposed change will reduce unnecessary regulatory burden and will allow AmerGen to take full advantage of the revisions to Title 10, Code of Federal Regulations (10

CFR), Section 50.72, ''Immediate notification requirements for operating nuclear power reactors,'' and 10 CFR 50.73, ''Licensee event report system.''

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

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

The proposed change involves an administrative change only. The proposed change does not involve the modification of any plant equipment or affect plant operation. The proposed change will have no impact on any safety related structures, systems or components. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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?

The proposed change has no impact on the design, function or operation of any plant structure, system or component. The proposed change is administrative in nature and does not affect plant equipment or accident analyses. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there is no change being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by deletion of the reporting requirement that is adequately addressed elsewhere.

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 amendment involves 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.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: March 25, 2005.

Description of amendment request: The proposed change would revise Technical Specification Surveillance Requirement (SR) 3.6.1.3.8 to add a note excluding leakage through primary containment penetrations 1MC–101 and 1MC–102 from the secondary containment bypass leakage total specified in the SR.

Implementation of this proposed change will provide operational flexibility by allowing Clinton Power Station (CPS) to utilize the additional margin in the regulatory dose limit analysis that supports the implementation of the alternative source term.

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 adds a note excluding the leakage through the primary containment purge lines from the secondary containment bypass leakage based on separate analysis of these paths using the assumptions in the alternative source term (AST) revision to the loss of coolant accident (LOCA) analysis.

The proposed change does not require modification to the facility. The proposed change in secondary containment bypass leakage does not affect the operation of any facility equipment, the interface between facility systems, or the reliability of any equipment. In addition, secondary containment bypass leakage does not constitute an initiator of any previously evaluated accidents. Therefore, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The radiological consequences of the LOCA analysis using the primary containment purge line leakage as separate from the secondary containment bypass leakage, has been evaluated as part of the application of AST assumptions. The results conclude that the radiological consequences 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. 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 does not affect the design, functional performance or operation of the facility. No new equipment is being introduced and installed equipment is not being operated in a new or different manner. Similarly, the proposed change does not affect the design or operation of any structures, systems or components involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created. There are no set points at which protective or mitigative actions are initiated that are affected by this proposed action. No change is being made to procedures relied upon to respond to an off-normal event.

As such the proposed amendment will 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.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of set points to initiate alarms or actions. The proposed change adds a note excluding the leakage through the primary containment purge lines from the secondary containment bypass leakage based on separate analysis of these paths using the assumptions in the AST revision to the LOCA analysis. There is no change in the design of the affected systems, no alteration of the set points at which alarms or actions are initiated, and no change in plant configuration from original design.

The margin of safety is considered to be that provided by meeting the applicable regulatory limits. The AST analysis indicates that the doses following a LOCA remain within the regulatory limits, and therefore, there is not a significant reduction in a margin of safety. The AST analysis confirms 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, operation of CPS in accordance with the proposed change will 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 amendment involves 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.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: April 1, 2005.

Description of amendment request: The proposed changes would incorporate into the Technical Specifications (TSs) the Oscillation Power Range Monitor (OPRM) instrumentation that will be declared operable within 30 days after completion of the February 2006 refueling outage. The proposed changes would add TS Section 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' and would revise TS Sections 3.4.1, ''Recirculation Loops Operating,'' and 5.6.5, ''Core Operating Limits Report (COLR).'' In addition, the changes would insert a new TS section for the OPRM instrumentation, delete the current thermal-hydraulic instability administrative requirements, and add the appropriate references for the OPRM trip set points and methodology. Clinton Power Station (CPS) will activate the automatic reactor protection system (*i.e.*, scram) outputs of the OPRM instrumentation upon implementation of these proposed TS changes.

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

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

Response: No.

The proposed changes specify limiting conditions for operation, required actions and surveillance requirements for the OPRM system, and allows operation in regions of the power to flow map currently restricted by the requirements of the Interim Corrective Actions (ICAs) and certain limiting conditions of operation of TS Section 3.4.1, ''Recirculation Loops Operating.'' The restrictions of the ICAs and TS Section 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of thermal-hydraulic oscillations that may develop into a thermal-hydraulic instability event. A thermal-hydraulic instability event has the potential to challenge the Minimum Critical Power Ratio (MCPR) safety limit. The OPRM system can automatically detect and suppress conditions necessary for thermal-hydraulic instability. With the activation of the OPRM system, the restrictions of the ICAs and TS Section 3.4.1 will no longer be required.

This proposed change has no impact on any of the existing neutron monitoring functions. When the OPRM is operable with operating limits as specified in the Core Operating Limits Report (COLR), the OPRM can automatically detect the imminent onset of local power oscillations and generate a trip signal. Actuation of a Reactor Protection System (RPS) trip (*i.e.*, scram) will suppress conditions necessary for thermal-hydraulic instability and decrease the probability of a thermal-hydraulic instability event. In the event the trip capability of the OPRM is not maintained, the proposed changes limit the period of time before an alternate method to detect and suppress thermal-hydraulic oscillations is required. CPS intends to utilize the ICAs as the alternative method for ensuring thermal-hydraulic oscillations do not occur. Since the duration of this period of time is limited, the increase in the probability of a thermal-hydraulic instability event is not significant.

Activation of the OPRM scram function will replace the current methods that require operators to insert an immediate manual reactor scram in certain reactor operating regions where thermal hydraulic instabilities could potentially occur. While these regions will continue to be avoided during normal operation, certain transients, such as a reduction in reactor recirculation flow, could place the reactor in these regions. During these transient conditions, with the OPRM instrumentation scram function activated; an immediate manual scram will no longer be required. This may potentially cause a marginal increase in the probability of occurrence of an instability event. This potential increase in probability is acceptable because the OPRM function will automatically detect the instability condition and initiate a reactor scram before the Minimum Critical Power Ratio (MCPR) Safety Limit is reached. Consequences of the potential instability event are reduced because of the more reliable automatic detection and suppression of an instability event, and the elimination of dependence on the manual operator actions. Operators monitor for indications of thermal hydraulic instability when the reactor is operating in regions of potential instability as a backup to the OPRM instrumentation.

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

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

Response: No.

The proposed changes replace procedural actions that were established to avoid operating conditions where reactor instabilities might occur with an NRC approved automatic detect and suppress function (*i.e.*, OPRM).

Potential failures in the OPRM trip function could result in either failure to take the required mitigating action or an unintended reactor scram. These are the same potential effects of failure of the operator to take the correct appropriate action under the current procedural actions. The effects of failure of the OPRM equipment are limited to reduced or failed mitigation, but such failure cannot cause an instability event or other type of accident.

The OPRM system uses input signals shared with the Average Power Range Monitor (APRM) system and rod block functions to monitor core conditions and generate a Reactor Protection System (RPS) trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of the other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. No new failure modes of either the new OPRM equipment or of the existing APRM equipment have been introduced.

Operation in regions currently restricted by the ICAs and TS Section 3.4.1 is within the nominal operating domain and ranges of plant systems and components for which postulated equipment and accidents have been evaluated. Therefore, operation within these regions does not create the possibility of a new or different kind of accident from any previously evaluated.

These proposed changes which specify limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allow operation in certain regions of the power-to-flow map 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 OPRM system monitors small groups of Local Power Range Monitor (LPRM) signals for indication of local variations of core power consistent with thermal-hydraulic oscillations and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated thermal-hydraulic instability event has the potential to result in a challenge to the MCPR safety limit. The OPRM system provides the capability to automatically detect and suppress conditions that might result in a thermal-hydraulic instability event and thereby maintains the margin of safety by providing automatic protection for the MCPR safety limit while reducing the burden on the control room operators significantly. The OPRM trip provides a trip output of the same type as currently used for the APRM. Its failure modes and types are similar to those for the present APRM output. Since the MCPR Safety Limit will not be exceeded as a result of an instability event following implementation of the OPRM trip function, it is concluded that the proposed change does not reduce the margin of safety.

Operation in regions currently restricted by the requirements of the ICAs and TS Section 3.4.1 is within the nominal operating domain assumed for identifying the range of initial conditions considered in the analysis of anticipated operational occurrences and postulated accidents. Therefore, operation in these regions does not involve a significant reduction in the margin of safety.

The proposed changes, which specify limiting conditions for operations, required actions and surveillance requirements of the OPRIVI system and allow operation in certain regions of the power to flow map, 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 amendment involves 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.

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

Date of amendment request: February 25, 2005.

Description of amendment request: The proposed change would delete Section 2.E of the Oyster Creek's Facility Operating License (FOL), DPR–16, which requires AmerGen Energy Company, LLC, to report violations of the requirements contained in Section 2.C of this license. The proposed change will reduce unnecessary regulatory burden and will allow AmerGen to take full advantage of the revisions to Title 10, Code of Federal Regulations (10 CFR), Section 50.72, ''Immediate notification requirements for operating nuclear power reactors,'' and 10 CFR 50.73, ''Licensee event report system.''

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

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

Response: No.

The proposed change involves an administrative change only. The proposed change does not involve the modification of any plant equipment or affect plant operation. The proposed change will have no impact on any safety related structures, systems or components. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

Response: No.

The proposed change has no impact on the design, function or operation of any plant structure, system or component. The proposed change is administrative in nature and does not affect plant equipment or accident analyses. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

Response: No.

The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there is no change being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by deletion of the reporting requirement that is adequately addressed elsewhere.

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 amendment involves 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: Richard J. Laufer.

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

Date of amendment request: March 17, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification 3.4.10, ''Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits,'' to replace the combination figure with separate P/T limit figures for each one of the three categories of operation: hydrostatic pressure test [Curve A], non-nuclear heatup and cooldown [Curve B], and nuclear (core critical) operation [Curve C]. The new curves also provide composite limits for all reactor pressure vessel (RPV) regions including core beltline region. RPV bottom head individual limit curves are superimposed on Curves A and B. In addition, two sets of curves are calculated; one for 32 effective full power years (EFPY) which represents the end of the current 40-year plant license and the other one is for 24 EFPY which has been selected as an intermediate point between the current EFPY and 32 EFPY.

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 revised P/T curves are based on the 1998 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, including the 2000 Addenda. This edition of the Code has been approved for use in both 10 CFR 50.55a and Regulatory Guide (RG) 1.147. The revised curves are also based on updated fluence calculations performed utilizing NRC-approved methodology consistent with RG 1.190 for calculating Reactor Pressure Vessel (RPV) neutron fluence. Revised fluence calculations are applicable for 24 and for 32 Effective Full Power Years (EFPY). The 32 EFPY represents a conservative exposure level at the end of the current 40-year plant operating license. The proposed change incorporates adjustment of the reference temperature for all beltline material to account for irradiation effects and provide a comparable level of protection as previously evaluated and approved. The adjusted reference temperature calculations were performed in accordance with the requirements of 10 CFR 50 Appendix G using the guidance contained in RG 1.99, Revision 2, to provide operating limits for up to 32 EFPY.

There are no changes being made to the RCS pressure boundary or to RCS material, design or construction standards. The proposed P/T curves define limits that continue to ensure the prevention of nonductile failure of the RCS pressure boundary. The revision of the P/T curves does not alter any assumptions previously made in the radiological consequence evaluations since the integrity of the RCS pressure boundary is unaffected. Therefore, the proposed changes will not significantly increase 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 revised P/T curves are based on a later edition and addenda of the ASME Code that incorporates current industry standards for the curves. The revised curves are also based on an RPV fluence that has been recalculated in accordance with the methodology of RG 1.190. The proposed change does not involve a modification to plant structures, systems or components. There is no effect on the function of any plant system, and no newly introduced system interactions. The proposed change does not create new failure modes or cause any systems, structures or components to be operated beyond their design bases. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed P/T curves define the limits of operation to prevent nonductile failure of the RPV upper vessel, bottom head and beltline region. The new curves conform to the guidance contained in RG 1. 190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' and RG 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials,'' and maintain the safety margins specified in 10 CFR 50 Appendix G. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 17, 2005. This amendment request supercedes, in its entirety, a previous application dated March 19, 2004, published in the **Federal Register** on June 22, 2004 (69 FR 34698).

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.6.1, ''Primary Containment Isolation Instrumentation,'' to correct a formatting error introduced during conversion to Improved Technical Specifications (ITS) by replacing ''1 per room'' with ''2'' for the required channels per trip system for the reactor water cleanup (RWCU) area ventilation differential temperature—high primary containment isolation instrumentation.

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

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

Response: No.

The proposed change restores the number of Required Channels Per Trip System of the RWCU Area Ventilation Differential Temperature—High isolation, Function 5.c of Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, to its pre-ITS value and adds a note to Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, that ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. No changes in operating practices or physical plant equipment are created as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change restores the number of Required Channels Per Trip System of the RWCU Area Ventilation Differential Temperature—High isolation, Function 5.c of Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, to its pre-ITS value and adds a note to Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, that ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. No physical change in plant equipment will result from this proposed change. Therefore, the proposed change does not create the possibility of a new or different type of accident from any previously evaluated.

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

Response: No.

The proposed change restores the number of Required Channels Per Trip System of the RWCU Area Ventilation Differential Temperature—High isolation, Function 5.c of Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, to its pre-ITS value and adds a note to Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, that ensures, during surveillance testing and normal operation, there will always be at least one instrument monitoring for a small leak in all RWCU locations. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: November 16, 2004.

Description of amendment request: The amendments would revise Technical Specifications (TS) 3.5.2, ''Emergency Core Cooling System,'' TS 3.6.6, ''Containment Spray System,'' TS 3.6.17, ''Containment Valve Injection Water System,'' TS 3.7.5, ''Auxiliary Feedwater System,'' TS 3.7.7, ''Component Cooling Water System,'' TS 3.7.8, ''Nuclear Service Water System (NSWS),'' TS 3.7.10, ''Control Room Area Ventilation System'' TS 3.7.12, ''Auxiliary Building Filtered Ventilation Exhaust System,'' and TS 3.8.1, ''AC Sources-Operating'' for Catawba, Units 1 and 2. The revisions would allow for the ''A'' and ''B'' NSWS headers to be take out of service for up to 14 days each for system upgrades.

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 operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The pipe repair project for the [nuclear service water system] NSWS and proposed [technical specifications] TS changes have been evaluated to assess their impact on normal operation of the systems affected and to ensure that the design basis safety functions are preserved. During the pipe repair the other NSWS train will be operable and no major maintenance or testing will be done on the operable train. The operable train will be protected to help ensure it would be available if called upon.

This pipe repair project will enhance the long term structural integrity in the NSWS system. This will ensure that the NSWS headers maintain their integrity to ensure its ability to comply with design basis requirements and increase the overall reliability for many years.

The increased NSWS train unavailability as a result of the implementation of this amendment does involve a one time increase in the probability or consequences of an accident previously evaluated during the time frame the NSWS headers are out of service for pipe repair. Considering this small time frame for the NSWS train outages with the increased reliability and the decrease in unavailability of the NSWS system in the future because of this project, the overall probability or consequences of an accident previously evaluated will decrease.

Therefore, because this is a temporary and not a permanent change, the time averaged risk increase is acceptable. The increase in the overall reliability of the NSWS along with the decreased unavailability in the future because of the pipe repair project will result in an overall increase in the safety of both Catawba units. Therefore, the consequences of an accident previously evaluated remains unaffected and there will be minimal impact on any accident consequences.

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

Response: No.

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed temporary TS changes do not affect the basic operation of the [emergency core cooling system] ECCS, [containment spray system] CSS, [containment valve injection water system] CVIWS, NSWS, [auxiliary feedwater] AFW, [component cooling water] CCW, [control room area ventilation system] [sic] CRAVS, [auxiliary building filtered ventilation exhaust system] ABFVES, or [emergency diesel generator] EDG systems. The only change is increasing the required action time frame from 72 hours (ECCS, CSS, NSWS, AFW, CCW, and EDG) or 168 hours (CVIWS, CRAVS and ABFVES) to 336 hours. The train not undergoing maintenance will be operable and capable of meeting its design requirements. Therefore, only the redundancy of the above systems is affected by the extension of the required action to 336 hours. During the project, contingency measures will be in place to provide additional assurance that the affected systems will be able to complete their design functions.

No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant, which will introduce any new accident causal mechanisms.

3. Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed temporary TS amendment. During the NSWS train outages, the affected systems will still be capable of performing their required functions and contingency measures will be in place to provide additional assurance that the affected systems will be maintained in a condition to be able to complete their design functions. No safety margins will be impacted.

The probabilistic risk analysis conducted for this proposed amendment demonstrated that the [core damage probability] CDP associated with the outage extension is judged to be acceptable for a one-time or rare evolution. Therefore, there is not 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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 8, 2005.

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

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. Will operation of the facility in accordance with this proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment introduces no new mode of plant operations and does not affect Structures, Systems, and Components (SSCs) associated with power production, accident mitigation, or safe plant shutdown. The SSCs affected by this proposed amendment are the Fuel Building Cask Handling Crane (FBCHC), the spent fuel storage canister, the spent fuel transfer cask, and the spent fuel inside the storage canister. A hypothetical 30 ft. drop of a loaded spent fuel shipping cask from the FBCHC is part of the River Bend Station (RBS) current licensing basis. With the proposed spent fuel transfer cask design and procedural changes implemented, the FCHC will be used to lift and handle a fuel-loaded spent fuel transfer cask of the same maximum weight and approximately the same dimensions as previously evaluated in the RBS USAR. The proposed amendment involves the use of redundant crane rigging during most lateral moves with a loaded spent fuel transfer cask, which provides temporary single-failure proof design features to provide protection against an uncontrolled lowering of the load or load drop. In those cases where the spent fuel transfer cask is not supported with redundant rigging, certain hypothetical, non-mechanistic load drops have been postulated and evaluated, with due consideration of the use of impact limiters in some locations.

With this amendment, the probability of a loaded spent fuel transfer cask drop is actually less likely than previously evaluated because the capacity of the spent fuel multi-purpose canister [MPC] (68 fuel assemblies) is larger than the capacity of the shipping cask described in the current licensing basis (18 fuel assemblies), which means that fewer casks will be required to be loaded, lifted, and handled for a given population of spent fuel assemblies. The consequences of the hypothetical spent fuel transfer cask load drops on plant SSCs are bounded by those previously evaluated for a shipping cask. That is, there is no significant damage to the Fuel Building structure or any SSCs used for safe plant shutdown. New analyses of hypothetical drops of a loaded transfer cask or canister confirm that there is no release of radioactive material from the storage canister and no unacceptable damage to the fuel, MPC, or transfer cask.

The hypothetical drop of a spent fuel canister lid into an open, fuel-filled canister in the spent fuel pool during fuel loading has also been evaluated. Again, this hypothetical accident is no more likely to occur than previously considered due to the higher capacity of the spent fuel transfer cask over the spent fuel shipping cask (*i.e.*, fewer casks will need to be loaded for a given number of fuel assemblies). The radiological consequences of this event due to the potential damage of spent fuel assemblies in the canister onto which the lid could be dropped have been evaluated. While more total fuel assemblies could potentially be damaged from a spent fuel canister lid drop compared to that assumed for the fuel handling accident described in the RBS current licensing basis, the significantly longer decay time of the spent fuel assemblies in the canister results in a much smaller source term, such that the existing fuel handling accident described in USAR Section 15.7.4 provides a bounding evaluation for the radiological consequences MPC lid drop. There is no rearrangement of the fuel or deformation of the fuel basket in the canister such that a critical geometry is created as a result of an MPC lid drop.

The likelihood of a spent fuel canister lid drop due to the failure of a crane component due to overload is very unlikely because the rated load of the crane (250,000 lbs) is approximately 16 times the weight of components lifted to install the canister lid.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment introduces no new mode of plant operations and does not affect SSCs associated with power production, accident mitigation, or safe plant shutdown. The SSCs affected by this proposed amendment are the non-single-failure-proof FBCHC, the spent fuel canister, the spent fuel transfer cask, and the spent fuel inside the canister. The design function of the FBCHC is not changed. The proposed amendment does not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators. The proposed amendment creates a new initiator of two accidents previously evaluated and caused by the non-mechanistic single failure of a component in the FBCHC load path.

The current licensing basis accidents for which new initiators are created by this amendment are the spent fuel shipping cask drop and the fuel handling accident. The RBS current licensing basis includes evaluations of the consequences of a spent fuel shipping cask drop and the consequences of the drop of a spent fuel assembly into the reactor core shortly after shutdown and reactor head removal. The new initiators include the drop of a spent fuel transfer cask of the same maximum weight and approximately the same dimensions as the shipping cask, and the drop of a spent fuel canister lid into an open, fuel filled canister in the spent fuel pool. Both of these new initiators create hypothetical accidents that are comparable in consequences to those previously evaluated. For the drop of a spent fuel transfer cask, the consequences are bounded by the current licensing basis analysis of the spent fuel shipping cask drop. That is, there is no significant damage to the Fuel Building structure or any SSCs used for safe plant shutdown, and there is no release of radioactive material. New analyses of the drop of a loaded transfer cask confirm that there is no release of radioactive material from the storage canister and no unacceptable damage to the fuel, MPC, or transfer cask.

For the drop of the spent fuel canister lid, the significantly longer decay time of the spent fuel assemblies in the canister compared to a spent fuel assembly in a recently shutdown reactor results in doses to the public that are less than the previously analyzed fuel handling accident. There is no rearrangement of the fuel in the canister such that a critical geometry is created as a result of an MPC lid drop.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment introduces no new mode of plant operations and does not affect SSCs associated with power production, accident mitigation, or safe plant shutdown. The SSCs affected by this proposed amendment are the non-single-failure-proof FBCHC, the spent fuel storage canister, the spent fuel transfer cask, and the spent fuel inside the canister. Therefore, this amendment does not affect the reactor or fuel during power operations, the reactor coolant pressure boundary, or primary or secondary containment. All activities associated with this amendment occur in the Fuel Building or in the adjacent outdoor truck bay area. The design function of the FBCHC is not changed. The proposed changes to plant operating procedures needed to implement dry spent fuel storage at RBS do not exceed or alter a design basis or safety limit associated with plant operation, accident mitigation, or safe shutdown. The FBCHC is used to lift and handle the spent fuel canister lid over spent fuel in the canister while in the spent fuel pool, and to lift and handle the spent fuel transfer cask, both when it is empty and after it is loaded with spent fuel in the spent fuel pool.

This proposed amendment results in a net safety benefit because a larger capacity cask is being used to move spent fuel out of the spent fuel pool that was previously evaluated (68 fuel assemblies versus 18 fuel assemblies), while maintaining the same maximum analyzed cask weight described in the USAR. This yields fewer casks to be loaded, fewer heavy load lifts, and, as a result, fewer opportunities for events such as load drops. Because the maximum weight of the loaded spent fuel transfer cask is the same as that assumed for the shipping cask and for which the FBCHC was designed, all design safety margins for use of the FBCHC remain unchanged. The rated capacity of the FBCHC is approximately 16 times that of components lifted to place the spent fuel canister lid, yielding significant safety margins for that particular lift.

Based on the above review, it is concluded that: (1) the proposed amendment does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed amendment; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Impact Statement.

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Allen G. Howe.

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

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

Docket No. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

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

Date of amendment request: February 25, 2005.

Description of amendment request: The proposed change would delete the applicable sections of the Facility Operating Licenses (FOLs); NPF–72, NPF–77, NPF–37, NPF–66, DPR–19, NPF–11, and NPF–18, respectively; which require Exelon Generation Company, LLC, to report violations of the requirements contained in Section 2.C of the Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 FOLs; Section 2.C of the Dresden Nuclear Power Station, Unit 2, renewed FOL; and Sections 2.C and 2.E of the LaSalle County Station, Units 1 and 2, FOLs. The proposed change will reduce unnecessary regulatory burden and will allow Exelon to take full advantage of the revisions to Title 10, Code of Federal Regulations (10 CFR), Section 50.72, ''Immediate notification requirements for operating nuclear power reactors,'' and 10 CFR 50.73, ''Licensee event report system.''

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

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

Response: No.

The proposed change involves an administrative change only. The proposed change does not involve the modification of any plant equipment or affect plant operation. The proposed change will have no impact on any safety related structures, systems or components. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

accident from any accident previously evaluated?

Response: No.

The proposed change has no impact on the design, function or operation of any plant structure, system or component. The proposed change is administrative in nature and does not affect plant equipment or accident analyses. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

Response: No.

The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there is no change being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by deletion of the reporting requirement that is adequately addressed elsewhere.

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

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: February 25, 2005.

Description of amendment request: The proposed change would delete the applicable sections of the Limerick Generating Station, Units 1 and 2, Facility Operating Licenses (FOLs), NPF–39 and NPF–85, which require Exelon Generation Company, LLC, (Exelon), to report violations of the requirements contained in Section 2.C of these licenses. The proposed change will reduce unnecessary regulatory burden and will allow AmerGen to take full advantage of the revisions to Title 10, Code of Federal Regulations (10 CFR), Section 50.72, ''Immediate notification requirements for operating nuclear power reactors,'' and 10 CFR 50.73, ''Licensee event report system.''

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

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

Response: No.

The proposed change involves an administrative change only. The proposed change does not involve the modification of any plant equipment or affect plant operation. The proposed change will have no impact on any safety related structures, systems or components. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

Response: No.

The proposed change has no impact on the design, function or operation of any plant structure, system or component. The proposed change is administrative in nature and does not affect plant equipment or accident analyses. The reporting requirement section of the FOL is not required because the requirements are either adequately addressed by 10 CFR 50.72 and 10 CFR 50.73, or other regulatory requirements, or are not required based on the nature of the Condition.

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

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

Response: No.

The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there is no change being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by deletion of the reporting requirement that is adequately addressed elsewhere.

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 amendment involves 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: Darrell J. Roberts.

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

Date of amendment request: February 11, 2005.

Description of amendment request: The proposed changes would modify the BVPS–1 and 2 Technical Specifications (TSs) to implement the relaxed axial offset control (RAOC) and F_Q surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control TSs, and increase the ability to return to power after a plant trip while still maintaining margin to safety limits under all operating conditions.

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

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

Response: No. The proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not initiate an accident. Evaluations and analyses of accidents, which are potentially affected by the parameters and assumptions, associated with the RAOC and $F_Q(Z)$ methodologies have shown that all design standards and applicable safety criteria will continue to be met. The consideration of these changes does not result in a situation where the design, material, or construction standards that were applicable prior to the change are altered. Therefore, the proposed changes will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The actual plant configuration, performance of systems, or initiating event mechanisms are not being changed as a result of the proposed changes. The design standards and applicable safety criteria limits will continue to be met, therefore, fission barrier integrity is not challenged. The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies have been shown not to adversely affect the plant response to postulated accident scenarios. The proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed changes do not challenge the performance or integrity of any safety-related system. The possibility for a new or different type of accident from any accident previously evaluated is not created since the proposed change does not result in a change to the design basis of any plant structure, system or component. Evaluation of the effects of the proposed changes has shown that all design standards and applicable safety criteria continue to be met.

Equipment important to safety will continue to operate as designed and component integrity will not be challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The proposed changes will not result in conditions that are more adverse and will not result in any increase in the challenges to safety systems.

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

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

Response: No. The proposed changes will not involve a significant reduction in a margin of safety. The proposed changes will assure continued compliance within the acceptance limits previously reviewed and approved by the NRC for RAOC and $F_Q(Z)$ methodologies. All of the appropriate acceptance criteria for the various analyses and evaluations will continue to be met.

The impact associated with the implementation of RAOC on peak cladding temperature (PCT) has been evaluated for the planned extended power uprate. This evaluation has determined that implementation of RAOC at the extended power uprate power level will not result in a significant reduction in a margin of safety for either unit.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: February 17, 2005.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7.7.1, ''Control Room Emergency Habitability Systems'' (BVPS–1), and TS 3.7.7, ''Control Room Emergency Air Cleanup and Pressurization System'' (BVPS–2), by dividing each specification into two specifications, addressing control room emergency ventilation and control room air cooling functions separately. Other minor changes are proposed to improve consistency with the Standard TSs and consistency between BVPS–1 and 2.

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

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

No.

The proposed changes do not adversely affect accident initiators or precursors or alter the design assumptions, conditions or configuration of the facility. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TSs for the control room ventilation systems which are mitigating systems designed to minimize inleakage, to filter the control room atmosphere and to provide heat removal for the control room envelope. These functions maintain the control room temperature within design limits and protect the control room personnel following accidents previously analyzed. The proposed changes do not alter or reduce the capability of the affected systems to maintain the control room temperature and protect the control room personnel consistent with the assumptions of the applicable safety analyses. Therefore, the probability of any accident previously evaluated is not significantly increased. The proposed change continues to assure [that] adequate system and component testing is performed to verify the operability of the control room habitability systems to ensure mitigation features are capable of performing the assumed functions. Therefore, the consequences of any accident previously evaluated are not significantly increased.

Therefore, 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 change create the possibility of a new or different kind of accident from any accident previously evaluated?

No.

The proposed changes will not adversely impact the accident analysis. The changes will not alter the requirements of the control room ventilation systems or their functions during accident conditions. No new or different accidents result from the application of the revised TS requirements. 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 significant change in the methods governing normal plant operation. The changes do not alter assumptions made in the safety analyses. The proposed changes are consistent with the safety analyses assumptions and current plant operating practices.

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?

No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed changes do 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 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: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

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: February 28, 2005.

Description of amendment request: The proposed amendments would allow the use of the Small Break Loss of Coolant Accident (SBLOCA) methodology described in Westinghouse WCAP 10054–P–A Addendum 2 Revision 1, ''Addendum to the Westinghouse Small Break emergency core cooling system (ECCS) Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model'' dated July 1997. This revised methodology determines the core response following a SBLOCA event and will be used to assure compliance with the post Loss of Coolant Accident (LOCA) acceptance criteria specified in 10 CFR 50.46.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054–P–A Addendum 2 Revision 1, ''Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model''.

The methodology used to perform small break loss of coolant accident (SBLOCA) analyses is not an accident initiator, thus changing the methodology does not increase the probability of an accident.

The fuel heat-up results generated by the proposed methodology will be utilized to demonstrate that the loss of coolant accident (LOCA) criteria for design basis for fission product barriers as described in 10 CFR Part 50.46 are not exceeded. The proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a Small Break LOCA, thus radioactive releases due to a SBLOCA accident are not affected by the proposed change in analysis methodology. Therefore, this change does not increase the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054–P–A Addendum 2 Revision 1, ''Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model''.

The analysis of a SBLOCA accident using the proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a Small Break LOCA.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the approved NOTRUMP SBLOCA Evaluation Model described in Westinghouse WCAP 10054–P–A Addendum 2 Revision 1, ''Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model''.

The methodology in the proposed licensing basis change has previously been reviewed and approved by the Nuclear Regulatory Commission as a conservative methodology. The Prairie Island configuration is representative of the modeling used in the methodology. Therefore, the proposed licensing basis change will result in a conservative calculation of fuel conditions following a SBLOCA event. This will ensure that there is no reduction in the margin of safety for Prairie Island SBLOCA analyses that utilize this methodology.

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

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

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 31, 2005.

Description of amendment request: The proposed amendment will increase the licensed power level to 1522 megawatts thermal (MWt) or 1.50 percent greater than the current power level of 1500 MWt. The requested increase in licensed rated power is the result of a measurement uncertainty recapture (MUR) power uprate. The information provided in support of this request is based on the NRC's Regulatory Issue Summary 2002–03, ''Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,'' dated January 31, 2002.

On July 18, 2003, the licensee submitted, and the NRC subsequently approved, an MUR power uprate amendment to increase the licensed power level to 1524 MWt or 1.6 percent greater than the current level of 1500 MWt. Problems during implementation resulted in the submission of an exigent license amendment request (LAR), which returned the licensed power to its original level (1500 MWt). The current LAR references the analysis from the July 18, 2003 submittal.

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.

Response: No.

There are no changes as a result of the MUR power uprate to the design or operation of the plant that could affect system, component, or accident functions. All systems and components function as designed and the performance requirements have been evaluated and found to be acceptable.

The reduction in power measurement uncertainty allows for safety analyses to continue to be used without modification. This is because those safety analyses were performed or evaluated at 102% of 1500 MWt (1530 MWt) or higher. Analyses at these power levels support a core power level of 1522 MWt with a measurement uncertainty of 0.5%. Radiological consequences of USAR [Updated Safety Analysis Report] Chapter 14 accidents were assessed previously using the alternate source term methodology (Reference 10.2 [Agencywide Documents Access Management System accession number ML013410095]). These analyses were performed at 102% of 1500 MWt (1530 MWt) and continue to be bounding. Updated Safety

Analysis Report (USAR) Chapter 14 analyses and accident analyses continue to demonstrate compliance with the relevant accident analyses' acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) were evaluated at an uprated core power level of 1524 MWt and continue to comply with their applicable structural limits. These analyses also demonstrate the components will continue to perform their intended design functions. Changing the heatup and cooldown curves is based on uprated fluence values. This does not have a significant effect on the reactor vessel integrity. Thus, there is no significant increase in the probability of a structural failure of the primary loop components. The LBB [leak before break] analysis conclusions remain valid and the breaks previously exempted from structural consideration remain unchanged.

All of the NSSS [nuclear steam system supplier] systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/BOP [nuclear steam system supplier/balance of plant] interface systems were evaluated at 1522 MWt and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. 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.

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety related systems or component and does not challenge the performance or integrity of any safety related system. 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.

Response: No.

Operation at 1522 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a 2% power measurement uncertainty or at uprated core powers that exceed the MUR uprated core power. System and component analyses have been completed at the MUR uprated core power conditions. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been both reviewed and approved by the NRC, or are currently under review (the proposed Pressure-Temperature Limits Report). Therefore, the proposed change does not involve a significant reduction in 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: December 28, 2004.

Description of amendment requests: The proposed amendments would relocate reactor coolant system related cycle-specific parameters from the Technical Specifications to the Core Operating Limits Report.

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

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

The proposed changes are programmatic and administrative in nature, which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions. More specific requirements regarding the safety limits (*i.e.*, departure from nucleate boiling ratio limit and peak fuel centerline temperature limit) are being imposed in Technical Specification (TS) 2.1.1, ''Reactor Core SLs [Safety Limits],'' which replace the reactor core safety limits figure and are consistent with the values stated in the Final Safety Analysis Report Update (FSARU). The proposed changes remove cycle-specific parameters from TS 3.4.1 and relocate them to the Core Operating Limits Report (COLR), which do not change the plant design or affect system operating parameters. In addition, the minimum limit for reactor coolant system (RCS) total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameters. The removal of the cycle-specific parameters from the TS does not eliminate existing requirements to comply with the parameters.

The proposed changes to TS 5.6.5b to reference only the topical report number and title for three of the topical reports do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to these topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval.

Although the relocation of the cycle-specific parameters to the COLR would allow revision of the affected parameters without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameters could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameters. The differences would not be significant and would be bounded by the existing requirement of TS 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameters being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The FSARU accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements, ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the type or increase the amount of any effluents released offsite.

Therefore, the proposed changes do 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 proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameters from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no changes being made to the parameters within which the plant is operated, other than their relocation to the COLR. There are no set points affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

The proposed changes to reference only the topical report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval.

Relocation of cycle-specific parameters has no influence or impact on, nor does it contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameters will continue to be calculated using the NRC reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis, and operation within the core operating limits will continue.

Therefore, the proposed changes do not create 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 margin of safety is established through equipment design, operating parameters, and the set points at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor do they affect the way in which safety-related systems perform their functions. The set points at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameters to the COLR will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to operate within prescribed limits.

The development of cycle-specific parameters for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that the plant operates within cycle-specific parameters.

The proposed changes to reference only the topical report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing topical reports would allow the use of current NRC-approved topical reports to support limits in the COLR without having to submit a request for an amendment to the operating license. Implementation of revisions to topical reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: December 31, 2004.

Description of amendment requests: The proposed amendments would revise Technical Specification 3.4.10, ''Pressurizer Safety Valves'' to add a separate Action and associated Completion Times for one or more inoperable pressurizer safety valves for the condition where the valves are inoperable solely due to loop seal temperatures being outside of design limits.

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.

This proposed change revises Technical Specification (TS) 3.4.10, ''Pressurizer Safety Valves,'' to add a separate Action and associated Completion Times (CTs) for one or more inoperable pressurizer safety valves (PSV) for the condition where the valves are inoperable solely due to loop seal temperatures being outside of design limits. Currently, when a PSV is in such a condition, it is conservatively declared inoperable and TS 3.4.10 Condition A is entered which has a CT of 15 minutes. A CT of 15 minutes normally provides insufficient time for restoring a PSV loop seal temperature to within limits. The new Action will provide CTs of 12 hours for exceeding the high temperature limit and 24 hours (MODES 1 and 2) or 72 hours (MODES 3 and 4) for exceeding the low temperature limit. In addition, two new PSV loop seal temperature surveillance requirements are proposed to assist in assuring PSV operability.

Loop seals are provided in the PSV inlet piping to maintain PSV body temperature within vendor recommended limits. This prevents PSV seat leakage that can result from spring relaxation with increased temperature. However, the water in the loop seals must be maintained at or above a minimum temperature to allow it to flash to steam when a PSV lifts. Because of the low density and low mass flow rate, PSV steam relief imposes minimal loading on the discharge piping ensuring acceptable pipe stresses. However, if cooler water is maintained in the loop seals, it may not flash completely, and a water and steam mixture could be discharged when a PSV lifts. Because of the higher density and higher mass flow rate, PSV relief of water and steam could impose increased loading and could result in unacceptably high pipe stresses on the discharge piping which could render the PSVs inoperable and/or damage the discharge piping.

The concern with the PSV opening during liquid relief conditions or with the loop seal temperature outside design limits, is the ability to ensure the valve reseats properly and no leakage occurs after the valve closes. However, even under liquid relief conditions, PSVs are still capable of providing their required relief capacity.

Failure of the PSV to reseat following discharge would result in an unisolable reactor coolant system leak. The consequences of such a leak are bounded by existing Final Safety Analysis Report Update (FSARU) accident analyses. Probabilistic risk assessment methods and a deterministic analysis have been utilized to determine there is no significant increase in core damage frequency or large early release frequency.

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.

Failure of one or more PSVs to reseat following discharge would result in an unisolable reactor coolant system leak. The consequences of such a leak are bounded by existing FSARU accident analyses and no new failure modes are introduced.

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

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

The proposed change is based upon both a deterministic evaluation and a risk-informed assessment.

The deterministic evaluation concluded that even with the loop seal temperature outside of design limits, causing one or more PSVs to be declared inoperable, the PSVs would still lift on demand to perform their safety function. Failure of one or more PSVs to reseat following discharge, resulting in an unisolable reactor coolant system leak, is an event bounded by existing FSARU accident analyses.

The risk assessment performed to support this license amendment request concluded that the increase in plant risk is small and consistent with the NRC's Safety Goal Policy Statement, ''Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,'' **Federal Register**, Volume 60, p. 42622, August 16, 1995 and guidance contained in of Regulatory Guides (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998 and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' dated August 1998.

Together, the deterministic evaluation and the risk-informed assessment provide high assurance that the PSVs will meet their design requirements.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: March 11, 2005.

Description of amendment requests: The proposed amendment would modify Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and 5.6.10, ''Steam Generator (SG) Tube Inspection Report,'' to allow the use of the SG tube W star (W*) alternate repair criteria (ARC) on a permanent basis. The W* ARC allows axial primary water stress corrosion cracking indications in the Westinghouse explosive tube expansion (WEXTEX) region to remain in service if the indication is located below the bottom of the WEXTEX transition. In addition, TS 5.6.10.d for NRC notification requirements of the voltage-based ARC would be revised.

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.

Of the various accidents previously evaluated, the permanent use of the steam generator (SG) tube W star (W*) alternate repair criteria (ARC) only affects the steam generator tube rupture (SGTR) accident evaluation and the postulated main steam line break (MSLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this evaluation.

For the SGTR accident, the required structural margins of the SG tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the Westinghouse explosive tube expansion (WEXTEX) region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' margins against burst are maintained for both normal and postulated accident conditions.

WCAP–14797–P, Revision 2, defines a length, W*, of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure-induced forces (with applicable safety factors applied). The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side as offset at higher tubesheet elevations by bow of the tubesheet. The proposed changes do not affect other systems, structures, components, or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or MSLB accident.

The consequences of an SGTR accident are affected by the primary-to-secondary leakage flow during the accident. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking in the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. No leakage has been observed in any in situ test of W* indications to date. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

MSLB leakage is limited by leakage flow restrictions resulting from the crack and tubesheet that provide a restricted leakage path and also limit the degree of crack face opening compared to free span indications. The total leakage, that is, the combined leakage for all such tubes, plus the combined leakage developed by any other ARC and non-ARC degradation, is limited to less than the maximum allowable MSLB accident dose analysis leak rate limit, such that offsite dose is maintained less than the guideline value in Title 10 to the Code of Federal Regulations (10 CFR) Part 100 and control room dose is maintained less than the value in General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50. In addition, the editorial changes made to Technical Specifications 5.5.9 and 5.6.10 have no impact on the MSLB leakage [and the SGTR].

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

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

Response: No.

The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon continued implementation of the W* ARC.

Axial indications left in service shall have the upper crack tip below the top of the tubesheet (TTS) by at least the value of the nondestructive examination (NDE) uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the tubesheet secondary face, thereby minimizing the potential for free span cracking and demonstrating that an acceptable level of risk is maintained for tubes returned to service under W* ARC. This repair criterion is in addition to ensuring that the upper crack tip is located below the bottom of the WEXTEX transition by at least the NDE measurement uncertainty. Condition monitoring will verify that all tube cracks returned to service under W* ARC remain below the TTS, including an allowance for NDE uncertainty.

These changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

Response: No.

The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. RG 1.121 is used as the basis in the development of the W* ARC for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube-burst that are consistent with the requirements of Section III of the ASME Code.

For primarily axially oriented cracking located within the tubesheet, tubeburst is precluded due to the presence of the tubesheet. WCAP–14797–P, Revision 2, defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Application of the W* ARC will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining MSLB leakage due to indications within the tubesheet region provides for large margins between calculated and actual leakage values. In addition, the total leakage, including leakage due to use of other ARC, is maintained below the maximum allowable MSLB accident dose analysis leak rate limit, such that offsite dose is maintained less than the guideline value in 10 CFR Part 100 and control room dose is maintained less than the value in GDC 19. In addition, the editorial changes made to Technical Specifications 5.5.9 and 5.6.10 have no impact on the determination of MSLB leakage [and the SGTR].

Plugging of the SG tubes reduces the reactor coolant flow margin for core cooling. Continued implementation of W* ARC will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

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

Based on the above evaluation, PG&E [Pacific Gas and Electric Company] concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Robert Gramm.

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 amendment request: November 9, 2004.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) 3.8.4, ''DC Sources–Operating,'' 3.8.5, ''DC Sources–Shutdown,'' 3.8.6, ''Battery Cell Parameters,'' and add a new TS Section, 5.5.13, ''Battery Monitoring and Maintenance Program.'' These changes are consistent with Technical Specifications Change Traveler (TSTF) 360, Revision 1 to request new actions with increased completion times for an inoperable battery chargers and alternate battery charger testing criteria for limiting condition for operation (LCO) 3.8.4 and LCO 3.8.5.

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

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

No. The proposed changes restructure the Technical Specifications (TSs) for the DC Electrical Power Systems. The proposed changes add actions to specifically address battery charger inoperability. This change will rely upon the capability of providing the battery charger function by an alternate means (*e.g.*, a 125 volts direct current (VDC) portable battery charger or a 250 VDC portable battery charger) to justify the proposed Completion Times. The DC electrical power systems, including associated battery chargers, are not initiators to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). Operation in accordance with the proposed TS ensures that the DC electrical power systems are capable of performing functions as described in the FSAR. Therefore the mitigative functions supported by the DC Power Systems will continue to provide the protection assumed by the analysis.

The relocation of preventive maintenance surveillance, and certain operating limits and actions to a newly-created, licensee-controlled TS 5.5.13, ''Battery Monitoring and Maintenance Program,'' will not challenge the ability of the DC electrical power systems to perform their design functions. The maintenance and monitoring required by current TS, which are based on industry standards, will continue to be performed. In addition, the DC Power Systems are within the scope of 10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' which will ensure the control of maintenance activities associated with the DC electrical power systems. The integrity of fission product barriers, plant configuration, and operating procedures as described in the FSAR will not be affected by the proposed changes.

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

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

No. The proposed changes involve restructuring the TS for the DC electrical power systems. These changes will rely upon the capability of providing the battery charger function by an alternate means to justify the proposed completion times when a normal battery charger is inoperable. The DC electrical power systems, which include the associated battery chargers, are not initiators to any accident sequence analyzed in the FSAR. Rather, the DC electrical power systems are used to supply equipment used to mitigate an accident. These mitigative functions, supported by the DC electrical power systems are not affected by these changes and they will continue to provide the protection assumed by the safety analysis described in the FSAR. There are no new types of failures or new or different kinds of accidents or transients that could be created by these changes.

Therefore, the proposed 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?

No. The margin of safety is established through equipment design, operating parameters, and the set points at which automatic actions are initiated. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the set points at which protective actions are initiated. Sufficient DC electrical system capacity is ensured to support operation of mitigation equipment. The changes associated with the new Battery Maintenance and Monitoring Program will ensure that the station batteries are maintained in a highly reliable state. The use of spare battery chargers will increase the reliability of the DC electrical systems during periods of normal battery charger inoperability. The equipment fed by the DC electrical sources will continue to provide adequate power to safety related loads in accordance with analysis assumptions. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Richard J. Laufer.

Southern Nuclear Operating Company, Inc., Docket No. 50–364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: January 19, 2005.

Description of amendment request: The proposed amendments would revise the Updated Final Safety Analysis Report to allow the use of fire rated electrical cable for fire areas 2–013 and 2–042 in lieu of a one hour rated electrical cable raceway fire barrier enclosure as described by Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix R, Section III.G.2 for protection of safe shutdown circuits.

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

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

The proposed 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) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. This is a revision to the FSAR to use [mineral insulated] MI cable in fire areas 2–013 and 2–042. The MI cable has been tested to applicable requirements and the implementation design reflects the test results. Therefore, the probability of any accident previously evaluated is not increased. Equipment required to mitigate an accident remain capable of performing the assumed function. Therefore, the consequences of any accident previously evaluated are not increased.

Therefore, it is concluded that this change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed change will not alter the requirements or function for systems required during accident conditions. No new or different accidents result from implementing MI cable for fire areas 2–013 and 2–042. The MI cable has been tested to applicable requirements, and the implementation design reflects the test results. The use of MI cable is not a significant change in the methods governing normal plant operation. The proposed change is consistent with the safety analysis assumptions and current plant operating practice.

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

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without mitigating actions. The proposed change does not affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition.

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

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 24, 2005.

Brief description of amendments: These proposed changes would revise Technical Specification (TS) 3.3.1 entitled ''Reactor Trip System Instrumentation'' (RTS) and TS 3.3.2 entitled ''Engineered Safety Feature Actuation System Instrumentation'' (ESFAS) Required Action Notes to reflect the wording in Standard Technical Specifications (STS) for plants with bypass capability per TS Task Force Traveler 418, Revision 2.

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

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

Response: No.

[Westinghouse Topical Report] WCAP–14333 provided the technical justification for relaxing various RTS and ESFAS Instrumentation bypass test times, Completion Times, and Surveillance Frequencies located in TS 3.3.1 and 3.3.2. As such, the proposed changes do not represent a significant hazards consideration or present a reduction in the margin of safety.

The protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same Reactor Trip System (RTS) Instrumentation and Engineered Safety Feature Actuation (ESFAS) Instrumentation will continue to be used and remain unchanged. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the TS do not result in a condition where the design, material, and construction standards, which were applicable prior to these changes, are altered.

The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [final safety analysis report].

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

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

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

Response: No.

There are no hardware changes nor is there any change in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in physical alteration to any plant system nor will there be any change in the method by which any safety-related plant system performs its safety function.

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

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

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

Response: No.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

Redundant RTS and ESFAS trains are maintained and diversity, with regard to the signals that provide reactor trip and engineered safety features actuation, is also maintained. All signals are credited as primary or secondary and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis.

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

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

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

NRC Section Chief: Allen G. Howe.

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: March 1, 2005.

Description of amendment request: The proposed changes to the Technical Specifications (TS) would revise the frequency for the Trip Actuating Device Operational Test of the P–4 Interlock Function and add Mode 4 to the Applicability for TS 3.3.2.

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

1. Do 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]. These interlocks and the associated testing do not directly initiate an accident. The consequences of accidents previously evaluated in the UFSAR are not adversely affected by these proposed changes because the changes are made to accurately reflect the design of the ESFAS [Engineered Safety Features Actuation System] system. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of any safety-related systems. 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 are made to accurately reflect the design of the ESFAS system. The nominal actuation set points specified by the Technical Specifications and the safety analysis limits assumed in the transient and accident analysis are unchanged. 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: John A. Nakoski.

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

Date of amendment request: March 8, 2005.

Description of amendment request: The proposed changes would revise the auxiliary feedwater (AFW) operability requirements and add an AFW allowed outage time and required actions.

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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to the AFW pump and flowpath requirements, as well as the revision of AFW surveillances, does not increase the probability of accidents previously evaluated since the AFW System is not required to operate until after the occurrence of the previously evaluated accidents. The change does not impact any of the initiators of the accidents. The proposed change does not involve a significant increase in the consequences of an accident previously evaluated because the AFW System will continue to perform its intended safety function for these accidents. The operation of the AFW System with the revised required action statements and added surveillances continues to meet the applicable design criteria.

2. Create the possibility of a new or different type of accident from any accident previously identified.

The safety function of the AFW System continues to be the same and is met using the same equipment. The change does not involve any plant modifications and does not revise the design of the plant or the AFW System. Operation of the AFW System with the revised required action statements and revised surveillances continues to meet the applicable design criteria and is consistent with the Surry accident analyses. Therefore, the proposed change does not introduce any new failures that could create the possibility of a new or different kind of accident from any accident previously identified.

3. Involve a significant reduction in a margin of safety.

The revised requirements for the AFW pumps and flowpaths, as well as the revision of AFW surveillances, continue to assure that the margins of safety assumed in the accidents and transients that rely upon operation of the AFW System are maintained. The proposed required action statements appropriately place the plant in a safe condition for the circumstances being addressed. Therefore, this proposed revision does not affect 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: 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: John A. Nakoski.

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

Date of amendment request: March 17, 2005.

Description of amendment request: The proposed change would incorporate a license condition that would permit irradiation of the fuel assemblies to a lead rod average burnup of 62,000 MWD/MTU.

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 is not significantly increased.

For most of the accidents analyzed in the UFSAR [Updated Final Safety Analysis Report] (*e.g.*, LOCA [loss-of-coolant accident], Steam Line Break, etc.) the fuel design has no impact on the likelihood of initiation of an accident. Fuel performance is evaluated as a consequence of the accident. The only accident where the fuel design may have an impact on the likelihood of a Chapter 14 accident is the Fuel Handling Accident discussed in Chapter 14.4.1 of the Surry UFSAR. The activity being evaluated is a slight increase in the lead rod average burnup limit for the fuel assemblies. No change in fuel design or fuel enrichment will be required to increase the lead rod average burnup. The fuel rods at the extended lead rod average burnup will continue to meet the design limits with respect to fuel rod growth, clad fatigue, rod internal pressure and corrosion. Thus, there will be no impact on the capability to engage the fuel assemblies with the handling tools. Therefore, it is concluded that the change will not result in more than a minimal increase in the frequency of occurrence of any accident previously evaluated in the UFSAR. The impact of extending the lead rod average burnup to 62,000 MWD/MTU from 60,000 MWD/MTU on the Core Kinetics Parameter, Core Thermal-Hydraulics/DNBR [Departure from Nucleate Boiling Ratio], Specific Accident Considerations, and Radiological Consequences was considered. Based on the evaluation of these considerations, it is concluded that increasing the lead rod average burnup limit to 62,000 MWD/MTU will not result in a significant increase in the consequences of the accidents previously evaluated in the Surry UFSAR.

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

The fuel is the only component affected by the change in the burnup limit. The change does not affect the thermal hydraulic response to any transient or accident. The fuel rod design criteria [will] continue to be met at the higher burnup limit. Thus, the change does not create the possibility of an accident of a different type.

3. The margin of safety as defined in the Bases to the Surry Technical Specifications is not significantly reduced.

The operation of the Surry cores with a limited number of fuel assemblies with some fuel rods irradiated to a lead rod average burnup of 62,000 MWD/MTU will not change the performance requirements of any system or component such that any design criteria will be exceeded. The normal limits on core operation defined in the Surry Technical Specifications will remain applicable for the irradiation of the fuel to a lead rod average burnup of 62,000 MWD/MTU. Therefore, the margin of safety as defined in Bases to the Surry 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 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: 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: John A. Nakoski.

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.*

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: July 6, 2004, as supplemented by letters dated September 21, and December 23, 2004.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to allow a one-time change in the Appendix J, Type A, Containment Integrated Leak Rate Test from the required 10 years to 15 years.

Date of issuance: April 6, 2005.

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

Amendment No.: 285.

Facility Operating License No. DPR–65: The amendment revised the TSs.

*Date of initial notice in **Federal Register:*** February 1, 2005 (70 FR 5237). The September 21 and December 23, 2004, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 6, 2004, as supplemented by letter dated August 5, 2004.

Brief description of amendments: The amendments revised the Technical Specifications to allow a diesel generator battery to remain operable with no more than one cell less than 1.36 Volts DC on float charge.

Date of issuance: March 29, 2005.

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

Amendment Nos.: 221 and 216.

Renewed Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 14, 2004 (69 FR 55469). The supplement dated August 5, 2004 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 March 29, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 28, 2004.

Brief description of amendments: The amendments eliminate the technical specification requirements to submit monthly operating reports and annual occupational radiation exposure reports.

Date of issuance: March 31, 2005.

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

Amendment Nos.: 222 and 217.

Renewed Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 23, 2004 (69 FR 68182).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 2005.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: June 3, 2003, as supplemented by letters dated July 29 and December 7, 2004, and January 18, 2005.

Brief description of amendments: The amendments revise TS 3.6.14 to allow a pressurizer enclosure hatch between the upper and lower containment volumes to be open for up to 6 hours to facilitate inspections of components such as the power operated relief valve block valves.

Date of issuance: April 5, 2005.

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

Amendment Nos.: 228/210.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43383). The supplemental letters dated July 29 and December 7, 2004, and January 18, 2005, provided clarifying information that did not change the initial proposed no significant hazards consideration determinations.

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

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: September 20, 2004.

Brief description of amendments: The amendments deleted the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: April 4, 2005.

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

Amendment Nos.: 227 and 209.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 1, 2005 (70 FR 5239)

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

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: September 28, 2004.

Brief description of amendments: The amendments eliminate the technical specification requirements to submit monthly operating reports and annual occupational radiation exposure reports.

Date of issuance: March 31, 2005.

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

Amendment Nos.: 226 and 208.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 23, 2004 (69 FR 68182).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 2005.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: October 16, 2003, as supplemented by letters dated May 11, 2004, and January 10, 2005.

Brief description of amendments: The amendments revised the Technical Specification (TS) 3.4.9 and the associated Bases to change the minimum pressurizer heater capacity from 126 kW to 400 kW to correct a non-conservative TS associated with a pressurizer design-basis deficiency.

Date of Issuance: March 28, 2005.

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

Amendment Nos.: 343, 345, & 344.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 20, 2004 (69 FR 2740).

The supplements dated May 11, 2004, and January 10, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on January 20, 2004 (69 FR 2740).

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

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: September 20, 2004.

Brief description of amendments: The amendments delete the Technical Specifications associated with hydrogen monitors.

Date of Issuance: April 4, 2005.

Effective date: As of the date of issuance and shall be implemented within 60 days after completion of the Spring 2005 refueling outage for Unit 1.

Amendment Nos.: 344, 346 & 345.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5239).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 5, 2004.

Brief description of amendment: This amendment revises Technical Specification Section 5.5.12, ''Primary Containment Integrity,'' to allow a one-time extension of its Appendix J, Type A, Containment Integrated Leak Rate Test interval from the current 10-year interval to a proposed 15-year interval.

Date of issuance: April 12, 2005.

Effective date: April 12, 2005, and shall be implemented within 30 days.

Amendment No.: 191.

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

Date of initial notice in ***Federal Register:*** August 31, 2004 (69 FR 53102).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 2005.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: June 24, 2004.

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

Date of issuance: April 6, 2005.

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

Amendment No.: 226.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 26, 2004 (69 FR 62474).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 30, 2004.

Brief description of amendment: The amendment changes the frequency for Technical Specification surveillance requirement (SR) 3.1.4.2, which verifies each tested control rod scram time is within limits with reactor steam dome pressure ≥ 800 psig. Specifically, the SR frequency increases from 120 days to 200 days of cumulative operation in MODE 1 (power operation).

Date of issuance: April 5, 2005.

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

Amendment No.: 283.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5241).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 2, 2004.

Brief description of amendment: The amendment revised Technical Specification (TS) 4.5.B.2.2 to change the surveillance requirement frequency for air testing the drywell and suppression pool spray headers and nozzles from ''once per 5 years'' to ''following maintenance that could result in nozzle blockage.''

Date of issuance: April 12, 2005.

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

Amendment No.: 214.

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

Date of initial notice in ***Federal Register:*** December 21, 2004 (69 FR 76490).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 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 13, 2004.

Brief description of amendments: The amendments eliminate the requirements in Technical Specifications (TSs) associated with hydrogen recombiners, and hydrogen and oxygen monitors.

Date of issuance: April 13, 2005.

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

Amendment Nos.: 173 and 135.

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

Date of initial notice in ***Federal Register:*** June 8, 2004 (69 FR 32073).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 13, 2005.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: April 30, 2004.

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

Date of issuance: April 11, 2005.

Effective date: As of the date of issuance, to be implemented within 180 days.

Amendment Nos.: 252 and 255.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 12, 2004 (69 FR 60681).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 31, 2004.

Brief description of amendment: The amendment revised Technical Specification 3.4.1, ''Recirculation Loops Operating,'' associated with single recirculation loop operation by incorporating limits for the linear heat generation rate fuel thermal limit into the limiting condition for operation.

Date of issuance: March 31, 2005.

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

Amendment No.: 134.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 4, 2005 (70 FR 401).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 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: September 21, 2004.

Brief description of amendment: The amendment deletes the Technical Specifications associated with hydrogen monitors.

Date of issuance: April 5, 2005.

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

Amendment No.: 216.

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

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5245).

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: July 6, 2004, as supplemented January 27, 2005.

Brief description of amendment: The amendment relocates the calibration requirement of Table TS 4.1–1, Item 22, ''Accumulator Level and Pressure,'' and the surveillance requirements of Table TS 4.1–1, Item 25, ''Portable Radiation Survey Instruments,'' from the Technical Specifications to licensee-controlled documents.

Date of issuance: April 6, 2005.

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

Amendment No.: 182.

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

Date of initial notice in ***Federal Register:*** August 31, 2004 (69 FR 53112).

The supplement dated January 27, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 2005.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: July 23, 2004, as supplemented January 6, 2005.

Brief description of amendments: The amendments modified the Technical Specification (TS) definition OPERABLE with respect to requirements for availability of normal and emergency power. Additionally, required actions for shutdown power TSs were modified.

Date of issuance: April 1, 2005.

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

Amendment Nos.: 264 and 246.

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

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9983).

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

No significant hazards consideration comments received: Comments received were addressed in the Safety Evaluation dated April 1, 2005.

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

Date of application for amendment: July 26, 2004, as supplemented on March 7, 2005.

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

Date of issuance: April 13, 2005.

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

Amendment No.: 89.

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

Date of initial notice in ***Federal Register:*** October 12, 2004 (69 FR 60685). The supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 13, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 10, 2004.

Brief description of amendments: These amendments delete the Technical Specifications associated with hydrogen monitors.

Date of issuance: March 29, 2005.

Effective date: March 29, 2005, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2—194; Unit 3—185.

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

Date of initial notice in ***Federal Register:*** January 18, 2005 (70 FR 2896).

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

No significant hazards consideration comments received: No.

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

Dates of application for amendments: February 26 and April 28, 2008, as supplemented by letters dated July 8 and October 20, 2004.

Brief description of amendments: The amendments revised Technical Specification (TS) Section 5.6.6, Reactor Coolant System (RCS) Pressure Temperature Limits Report (PTLR), to facilitate future licensee-controlled changes to the PTLR. The changes include a revised PTLR that provides new heatup and cooldown limits and Cold Overpressure Protection System (COPS) set points, and to recalculate the minimum size of the pressurizer power operated relief valve orifice of the RCS vent. In addition, the changes relocate the COPS arming temperature to the PTLR, and lower the COPS arming temperature from 350 °F to 220 °F. The licensee also included TS bases changes to support the changes to the TSs.

Date of issuance: March 28, 2005.

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

Amendment Nos.: 136 (Unit 1) and 115 (Unit 2).

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 13, 2004 (69 FR 19575) and April 22, 2004 (69 FR 34707).

The supplements dated July 8 and October 20, 2004, 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 March 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 14, 2004.

Brief description of amendments: The amendments eliminate the technical specification requirements to submit monthly operating reports and annual occupational radiation exposure reports.

Date of issuance: April 5, 2005.

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

Amendment Nos.: 300 and 289.

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

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5250).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 8, 2004.

Brief description of amendment: The amendment eliminates the requirements in Technical Specifications to submit monthly operating reports and annual occupational radiation exposure reports.

Date of issuance: March 21, 2005.

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

Amendment No.: 57.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 18, 2005 (70 FR 2902).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 2005.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: September 8, 2004.

Brief description of amendment: These amendments delete the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: March 22, 2005.

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

Amendment Nos.: 238 and 219.

Renewed Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 18, 2005 (70 FR 2902).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 23, 2004.

Brief Description of amendments: These amendments revise the Technical Specifications Section 3.16, ''Emergency Power System,'' requirements for verifying the operability of the remaining emergency diesel generator (EDG) when either unit's dedicated EDG or the shared backup EDG is inoperable.

Date of issuance: April 5, 2005.

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

Amendment Nos.: 241 and 240.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 19, 2004 (69 FR 51490).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 21, 2004.

Brief Description of amendments: These amendments revise the Technical Specifications by eliminating the requirements to submit monthly operating reports and occupational radiation exposure reports.

Date of issuance: March 22, 2005.

Effective date: March 22, 2005.

Amendment Nos.: 240 and 239.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 18, 2005 (70 FR 2903).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 18th day of April 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

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