For the Nuclear Regulatory Commission.

Pao-Tsin Kuo,

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

[FR Doc. 04–8286 Filed 4–12–04; 8:45 am]

BILLING CODE 7590–01–U

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Nuclear Waste; Revised

The agenda for the 149th meeting of the Advisory Committee on Nuclear Waste (ACNW) scheduled for April 20–22, 2004, 11545 Rockville Pike, Rockville, Maryland, has been revised to include a presentation on the Scientific and Technical Priorities at Yucca Mountain on Wednesday, April 21, 2004, as follows:

4 p.m.–5 p.m.: Scientific and Technical Priorities at Yucca Mountain (Open)—The Committee will hear presentations by and hold discussions with representatives of the Electric Power Research Institute regarding their December 2003 report on scientific and technical priorities at Yucca Mountain.

All other items pertaining to this meeting remain the same as previously published in the **Federal Register** on Thursday, April 1, 2004 (69 FR 17243).

For further information, contact Mr. Howard J. Larson, Special Assistant, ACNW, (Telephone: 301–415–6805), between 7:30 a.m. and 4:15 p.m., ET.

Dated: April 7, 2004.

J. Samuel Walker,

Acting Secretary of the Commission.

[FR Doc. 04–8285 Filed 4–12–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of April 12, 19, 26, May 3, 10, 17, 2004.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of April 12, 2004

Tuesday, April 13, 2004

9:30 a.m. Briefing on Status of Office of Nuclear Regulatory Research (RES) Programs, Performance, and Plans (Public Meeting) (Contact: Alan Levin, 301–415–6656).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of April 19, 2004—Tentative

Therea re no meetings scheduled for the Week of April 19, 2004.

Week of April 26, 2004—Tentative

Wednesday, April 28, 2004

9:30 a.m. Discussion of Security Issues (Closed—Ex. 1)

Week of May 3, 2004—Tentative

Tuesday, May 4, 2004

9:30 a.m. Briefing on Results of the Agency Action Review Meeting (Public Meeting) (Contact: Bob Pascarelli, 301–415–1245).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Thursday, May 6, 2004

1:30 p.m. Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of May 10, 2004—Tentative

Monday, May 10, 2004

1:30 p.m. Briefing on Grid Stability and Offsite Power Issues (Public Meeting) (Contact: Cornelius Holden, 301–415–3036).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Tuesday, May 11, 2004

9:30 a.m. Briefing on Status of Office of International Programs (OIP) Programs, Performance, and Plans (Public Meeting) (Contact: Ed Baker, 301–415–2344).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

1:30 p.m. Briefing on Threat Environment Assessment (Closed—Ex. 1).

Week of May 17, 2004—Tentative

There are no meetings scheduled for the Week of May 17, 2004.

* The scheduled for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Dave Gamberoni, (301) 415–1651.

SUPPLEMENTARY INFORMATION: By a vote of 3–0 on April 1, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that "Discussion of Security Issues (Closed—Ex. 1)'' be held April 7, and on less than one week's notice to the public.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

* * * * *

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: April 8, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–8419 Filed 4–9–04; 9:24 am]

BILLING CODE 7590–01–M

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, March 19 through April 1, 2004. The last biweekly notice was published on March 30, 2004 (69 FR 16615).

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 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/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: February 27, 2004.

Description of amendment request: The licensee proposed to relocate the average power range monitor (APRM)-based stability protection settings for Option II stability solution to the Core Operating Limits Report (COLR). The Option II solution demonstrates that existing quadrant-based APRM trip systems will initiate a reactor scram for postulated reactor instability and avoid violating the minimum critical power ratio safety limit. Use of Option II was previously approved by the Nuclear Regulatory Commission staff thru Amendment No. 235, dated October 18, 2002.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee 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 relocate the Average Power Range Monitor (APRM) based stability protection settings for the Option II stability solution from the Technical Specifications (TS) to the Core Operating Limits Report (COLR). The APRM based stability protection settings are not an initiator or a precursor to an accident. Furthermore, changes to the stability protection settings do not physically modify or change the function, or system interfaces, of the APRM Neutron Flux Scram and Neutron Flux Control Rod Block systems or components. The APRM based stability protection settings provide automatic protection to assure that anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. The proposed TS changes cannot increase the consequences of a previously evaluated accident because the changes do not alter any Limiting Safety System Setting, but only relocate the applicable stability protection settings to the COLR. The applicable stability protection settings will continue to be determined by an NRC approved methodology.

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 will relocate the APRM based stability protection settings for the Option II stability solution from the TS to the COLR. The APRM based stability protection settings for the Option II stability solution assure anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. These changes do not introduce any new accident precursors and do not involve any alterations to plant configurations which could initiate a new or different kind of accident. The proposed changes do not affect the intended function of the APRM system nor do they affect the operation of the system in a way which would create a new or different kind of accident.

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

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

Response: No.

The proposed change will relocate the APRM based stability protection settings for the Option II stability solution from the TS to the COLR. The APRM based stability protection settings for protection against reactor instability assure anticipated coupled neutronic/thermal-hydraulic instabilities will not compromise established fuel safety limits. No fuel thermal limits or other design and licensing basis acceptance criteria are adversely affected. No other events are adversely affected. The margin of safety, as defined in the TS, for all events is maintained.

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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: March 8, 2004.

Description of amendment request: The proposed amendment would delete Operating License Condition 2.C.(6) "Long Range Planning Program.'' The original objective of this requirement was to enable the licensee to better control and manage resources regarding major activities. The license condition does not have any direct effect on plant design or operation. Since imposition of this requirement on May 27, 1988, the licensee has developed internal processes to control and manage work activities, thus leading the licensee to determine that this license condition is no longer needed.

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. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The subject license condition was not a factor in the scenario of any previously analyzed postulated design-basis accident or anticipated operational transient. No hardware design change is involved with the proposed amendment. Thus, the proposed deletion of the license condition would create no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the previously analyzed consequences of postulated accidents and anticipated operational transients. Accordingly, the deletion of the license condition will lead to no increase in the consequences of an accident previously evaluated, and no increase in the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods the unit is operated. As a result, all SSCs will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the proposed deletion of the license condition will not lead the licensee to exceed or alter a design basis or safety limit, and will not result in operating any component in a less conservative manner, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the NRC staff's analysis, 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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard J. Laufer.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: March 8, 2004.

Description of amendment request: The proposed amendment would delete Operating License Condition 2.C.(9) "Long Range Planning Program.'' The original objective of this requirement was to enable the licensee to better control and manage resources regarding major activities. The license condition does not have any direct effect on plant design or operation. Since imposition of this requirement on May 27, 1988, the licensee has developed internal processes to control and manage work activities, thus leading the licensee to determine that this license condition is no longer needed.

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. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The subject license condition was not a factor in the scenario of any previously analyzed postulated design-basis accident or anticipated operational transient. No hardware design change is involved with the proposed amendment. Thus, the proposed deletion of the license condition would create no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the previously analyzed consequences of postulated accidents and anticipated operational transients. Accordingly, the deletion of the license condition will lead to no increase in the consequences of an accident previously evaluated, and no increase in the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods the unit is operated. As a result, all SSCs will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the proposed deletion of the license condition will not lead the licensee to exceed or alter a design basis or safety limit, and will not result in operating any component in a less conservative manner, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the NRC staff's analysis, 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: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: December 12, 2003.

Description of amendments request: The proposed amendment would delete Technical Specification (TS) Section 5.5.3, "Post-Accident Sampling,'' requirements to maintain a Post-Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as a result of NUREG–0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power

Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the NRC's lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in a license amendment application in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated December 12, 2003.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

Therefore, this change does not involve a significant reduction in [a] margin of safety.

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

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

Attorney for licensee: James M. Petro, Jr., Esquire, Counsel, Constellation Energy Group, Inc., 750 East Pratt Street, 5th floor, Baltimore, MD 21202.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 25, 2003.

Description of amendment request: The proposed amendments would correct two inadvertent editorial changes made by Duke during the submittal of Technical Specification (TS) Amendment 194/175 which revised TS 3.3.1 (Reactor Trip System Instrumentation) and TS Amendment 197/178 which revised TS 4.2.1 (Design Features, Fuel Assemblies).

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 this LAR [License Amendment Request] involve a significant increase in the probability or consequences of an accident previously evaluated?

No. Approval and implementation of this LAR will have no affect on accident probabilities or consequences since the proposed changes are editorial in nature and were previously reviewed and approved by the NRC [Nuclear Regulatory Commission].

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

No. This LAR does not involve any physical changes to the plant. Therefore, no new accident causal mechanisms will be generated. The proposed changes are editorial in nature and were previously reviewed and approved by the NRC. Consequently, plant accident analyses will not be affected by these changes.

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

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be affected by the proposed changes since they are editorial in nature and have been previously reviewed and approved by the NRC.

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, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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 amendment request: January 15, 2004, as supplemented by letter dated March 15, 2004.

Description of amendment request: The proposed amendments would revise the Technical Specifications associated with the control rod drive (CRD) trip devices. These amendments are needed to support implementation of the reactor trip breaker (RTB) replacement.

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 LAR [license amendment request] modifies the Technical Specifications [TS] to incorporate new TS requirements associated with the new Control Rod Drive (CRD)/Reactor Trip Breaker (RTB) configuration. The proposed LAR will continue to ensure that the CRD trip devices will be operable to ensure that the reactor remains capable of being tripped at any time it is critical. Reliable CRD reactor trip circuit breakers and associated support circuitry provides assurance that a reactor trip will occur when initiated. The new RTBs will have the same seismic and quality group qualifications as the existing components in the CRDCS [CRD control system] system [sic]. The new RTBs will enhance the reliability of the system by resolving age-related degradation issues and replacing obsolete equipment. Therefore, the proposed LAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated[.]

The proposed LAR modifies the Technical Specifications to incorporate new TS requirements associated with the new CRD/RTB configuration. The systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. Rather, the systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes do not affect the mitigating function of these systems. The reliability of the mitigating systems will be improved by implementation of the RTB Upgrade. Consequently, these changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed TS changes do not unfavorably affect any plant safety limits, set points, or design parameters. The changes also do not unfavorably affect the fuel, fuel cladding, RCS [reactor coolant system], or containment integrity. Therefore, the proposed TS change, which adds TS requirements associated with the CRD/RTB upgrade, do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 3, 2004.

Description of amendment request: The proposed amendments would revise the administrative Technical Specifications (TSs) for the Reactor Coolant Pump Flywheel Inspection Program to extend the allowable inspection interval to 20 years.

The NRC staff issued a notice of opportunity for comment in the ***Federal Register*** on June 24, 2003 (68 FR 37590), on possible amendments to extend the inspection interval for reactor coolant pump (RCP) flywheels, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 22, 2003 (68 FR 60422). The licensee affirmed the applicability of the model NSHC determination in its application dated March 3, 2004.

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

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

The proposed change to the RCP flywheel examination frequency does not change the response of the plant to any accidents. The RCP will remain highly reliable and the proposed change will not result in a significant increase in the risk of plant operation. Given the extremely low failure probabilities for the RCP motor flywheel during normal and accident conditions, the extremely low probability of a loss-of-coolant accident (LOCA) with loss of offsite power (LOOP), and assuming a conditional core damage probability (CCDP) of 1.0 (complete failure of safety systems), the core damage frequency (CDF) and change in risk would still not exceed the NRC's acceptance guidelines contained in Regulatory Guide (RG) 1.174 ($<$1.0E–6 per year). Moreover, considering the uncertainties involved in this evaluation, the risk associated with the postulated failure of an RCP motor flywheel is significantly low. Even if all four RCP motor flywheels are considered in the bounding plant configuration case, the risk is still acceptably low.

The proposed change does not adversely affect accident initiators or precursors, nor alter the design assumptions, conditions, or configuration of the facility, or the manner in which the plant is operated and maintained; alter or prevent the ability of structures, systems, components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits; or affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the type or amount of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposure. The proposed change is consistent with the safety analysis assumptions and resultant consequences. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change in flywheel inspection frequency does not involve any change in the design or operation of the RCP. Nor does the change to examination frequency affect any existing accident scenarios, or create any new or different accident scenarios. Further, the change does not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or alter the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements, and does not alter any assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not 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 impacted by this change. The proposed change will not result in plant operation in a configuration outside of the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in RG 1.174. There are no significant mechanisms for inservice degradation of the RCP flywheel. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: March 9, 2004.

Description of amendment request: The proposed amendment would extend the completion time (CT) from 1 hour to 24 hours for Condition B of Technical Specification (TS) 3.5.1, "Accumulators.'' The accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. Condition B of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved Topical Report WCAP–15049–A, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated March 9, 2004.

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

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

The basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049–A is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049–A for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,'' and 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049–A, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049–A evaluation as a worst case data point. In addition, WCAP–15049–A states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049–A evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049–A evaluation demonstrates that the small increase in risk due to increasing the CT for an inoperable accumulator is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

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

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk.

Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049–A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049–A evaluation.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: December 24, 2003.

Description of amendment request: The proposed amendment would delete requirements in the Pilgrim Nuclear Power Station Technical Specifications (TSs) 3.7.A.7.c and 4.7.A.7.c, associated with hydrogen analyzers. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50374), on possible amendments to eliminate the hydrogen analyzers from TSs, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the Consolidated Line Item Improvement Process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the relevant portions of the model NSHC determination in its application dated December 24, 2003.

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

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

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

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG [Regulatory Guide] 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs [Severe Accident Management Guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

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

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

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

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

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

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

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

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

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

Based upon the reasoning presented above, the requested change does not involve a significant hazards consideration. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Darrell J. Roberts, Acting.

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

Date of amendment request: March 15, 2004.

Description of amendment request: The amendment proposes to move the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specification (TS) 3.4.8.2, pressurizer heatup and cooldown limits to the Technical Requirements Manual (TRM), which is reviewed in accordance with Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR), "Changes, tests, and experiments.'' The associated action statement, surveillance requirement, and bases are also proposed for relocation.

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

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

Response: No.

The probability of an accident is unchanged as a result of the proposed change to delete the Waterford 3 pressurizer heatup and cooldown rates and associated action, surveillance requirement, and bases from the TS. The cooldown and heatup rates are not initiators to any accidents or pressurizer transients discussed in the Waterford 3 Final Safety Analysis Report (FSAR). Therefore, the probability of an accident is not changed.

The purpose of the pressurizer heatup and cooldown limits is to ensure that given transient events will not negatively affect the pressurizer structural integrity beyond Code allowables. These limits will be maintained within ASME [American Society of Mechanical Engineers] Code allowables in the TRM in accordance with 10 CFR 50.59. Therefore, the consequences of an accident are not increased.

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

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

Response: No.

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for the flaw evaluation and fatigue analysis performed in accordance with the ASME Code Section XI, subsection IWB–3600 requirements. The Waterford 3 FSAR has analyzed the conditions that would result from a thermal or pressurization transient on the Waterford 3 pressurizer. The proposed deletion of the pressurizer heatup and cooldown rates and relocation of the limits to the TRM does not change the way that the pressurizer is designed or operated.

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

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

Response: No.

The margin of safety is established by the rules contained in the ASME Section III Code. Any future changes to the cooldown or heatup rates will be evaluated using 10 CFR 50.59 and are required to meet the ASME Code margins.

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

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

Attorney for licensee: N.S. Reynolds, Esquire, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 12, 2004.

Description of amendment request: The proposed amendments would modify the Technical Specifications (TS) to eliminate selected response time testing (RTT) requirements associated with Reactor Protection System instrumentation and Primary Containment Isolation instrumentation for Main Steam Line Isolation functions. The proposed changes are consistent with the Boiling Water Reactor Owners Group (BWROG) Licensing Topical Report "System Analyses for the Elimination of Selected Response Time Testing Requirements,'' NEDO–32291√A, Supplement 1, dated October 1999, as approved by the NRC on June 11, 1999.

The original Licensing Topical Report (LTR) NEDO–32291–A, dated October 1995, established a generic basis for elimination of many RTTs for instrument loops that had good performance histories and longer response time requirements. The justification was based on the adequacy of surveillance tests other than RTTs to assure that response time requirements were met for sensors in those loops. Supplement 1 to NEDO–32291–A was prepared to document an analysis to extend the conclusions of the original study to cover the logic components in selected instrumentation loops that have intermediate length response time requirements. The intent was to demonstrate that elimination of the RTT requirements for the logic portions of those loops is of no safety significance. Supplement 1 concludes, for instrument loops meeting the application criteria of the Licensing Topical Report, that performance of ongoing TS required surveillance tests other than RTTs (*i.e.*, calibration tests, functional tests, and logic system functional tests) provides adequate assurance that those instrument loops will meet their respective response time requirements.

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 probability or consequences of an accident previously evaluated.

The proposed amendment to the TS eliminates selected RTT requirements in accordance with the NRC approved BWROG LTR. Elimination of RTT for selected instrumentation in the Reactor Protection System and Primary Containment Isolation Instrumentation does not result in the alteration of the design, material, or construction standards that were applicable prior to the proposed change. The response time assumptions used in the accident analyses remain unchanged. Only the methodology used for response time verification is changed. All component models used in the affected trip channels were analyzed for a bounding response time. As documented in the BWROG LTR and supplement, a degraded response time will be detected by other TS required tests. The bounding response time of the relays discussed in the supplement to the LTR can be used in place of actual measured response times to ensure that the instrumentation systems will meet the response time requirements of the accident analysis.

The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. In addition, the proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident.

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

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

The proposed action does not involve physical alteration of the station. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which LaSalle is operated. There are no setpoints at which protective or mitigative actions are initiated that are affected by this proposed action. All Reactor Protection System and Primary Containment Isolation Instrumentation channels affected by the proposed change will continue to have an initial response time verified by test before initially placing the channel in service and after any maintenance that could affect response time.

The proposed change does not alter assumptions made in the safety analysis. A review of the failure modes of the affected sensors and relays indicates that a sluggish response of the instruments can be detected by other TS surveillances. Changing the method of periodically verifying instrument response for the selected instrument channels will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the channel characteristic. This proposed action will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

The sensors and relays in the affected channels will be able to meet the bounding response times as defined and presented in the LTR Supplement. It has been found acceptable to use component bounding response times in place of actual measured response times to ensure that instrumentation systems will meet response time requirements of the accident analyses. In addition, [Exelon Generation Company, LLC] EGC's adherence to the conditions listed in the NRC Safety Evaluations for the LTR and Supplement provides additional assurance that the instrumentation systems will meet the response time requirements of the accident analyses.

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

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

Implementation of the BWROG LTR methodologies for eliminating selected response time testing requirements does not involve a significant reduction in the margin of safety. The current response time limits are based on the maximum values assumed in the plant safety analyses. The analyses conservatively establish the margin of safety. The elimination of the selected response time testing does not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for plant safety analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore, the margin of safety is not affected. This is based on the ability to detect a degraded response time of an instrument or relay by the other required TS tests, component reliability, and redundancy and diversity of the affected functions, as justified in the reviewed and approved LTR and Supplement.

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. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Dockets 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: February 27, 2004.

Description of amendment request: The proposed change to the Technical Specifications (TSs) supports the activation of the trip outputs of the previously-installed Oscillation Power Range Monitor (OPRM) portion of the Power Range Neutron Monitoring (PRNM) system. Specifically, this proposed change will revise TS Sections 3.3.1.1, "Reactor Protection System Instrumentation,'' and 3.4.1, "Recirculation Loops Operating Reporting Requirements,'' and their associated TS Bases, and TS Section 5.6.5, "Core Operating Limits Report (COLR).'' In addition, the proposed change deletes the Interim Corrective Action requirements from the Recirculation Loops Operating TSs.

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

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

Response: No. This modification has no impact on any of the previously installed PRNM functions. Plant operation in portions of the former restricted region 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 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.

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 modification replaces procedural actions that were established to avoid operating conditions where reactor instabilities might occur with an NRC approved automatic detect and suppress function.

Potential failures in the OPRM Upscale 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 net effect of the modification changes the method by which an instability event is detected and by which mitigating action is initiated, but does not change the type of stability event that could occur. 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.

Therefore, since no radiological barrier will be challenged as a result of activating the OPRM trip function, it is concluded that this proposed activity does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The current safety analyses assume that the existing procedural actions are adequate to prevent an instability event. As a result, there is currently no quantitative or qualitative assessment of an instability event with respect to its impact on MCPR.

The OPRM trip function is being implemented to automate the detection (via direct measurement of neutron flux) and subsequent suppression (via scram) of an instability event prior to exceeding the MCPR Safety Limit. The OPRM trip provides a trip output of the same type as currently used for the Average Power Range Monitor (APRM). Its failure modes and types are identical to those for the present APRM output. Currently, the MCPR Safety Limit is not impacted by an instability event since the event is "mitigated'' by manual means via the procedural actions, which prevent plant operating conditions where an instability event is possible. In both methods of mitigation (manual and automated), the margin of safety associated with the MCPR Safety Limit is maintained.

Therefore, 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 involve a significant reduction in a margin of safety.

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

Attorney for Licensee: Mr. Edward Cullen, Vice President and General Counsel, Exelon Generation Company, LLC, 2301 Market Street, S23–1, Philadelphia, PA 19101.

NRC Section Chief: Darrell Roberts, Acting.

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

Date of amendment request: February 27, 2004.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) Section 5.6.2.6, "Post Accident Sampling,'' requirements to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the NRC's lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 3, 2003 (68 FR 10052) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in a license amendment application in the **Federal Register** on May 13, 2003 (68 FR 25664). The licensee affirmed the applicability of the following NSHC determination in its application dated February 27, 2004.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602–1551.

NRC Section Chief: William F. Burton, Acting.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: January 28, 2004.

Description of amendment request: Duane Arnold Energy Center implemented improved technical specifications in 1998 via Amendment 223 using NUREG 1433, "Standard Technical Specifications—General Electric Plants BWR/4,'' Revision 1, as a model. The proposed amendment would revise Technical Specification Sections 5.5.11, 1.4, 3.3.1.1, and 5.5.2 to adopt the following selected NRC approved generic changes to the improved technical specification NUREG.

- Technical Specification Task Force (TSTF)–273, Revision 2, Safety Function Determination Program Clarifications.
- TSTF–284, Revision 3, Add "Met'' versus "Perform'' to Specification 1.4, Frequency.
- TSTF–264, Deletion of Flux Monitors Specific Overlap Surveillance Requirements.
- TSTF–299, Administrative Controls Program 5.5.2.b Test Interval Defined and Allowance for 25 Percent Extension of Frequency.

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:

Adoption of TSTF–273, Revision 2, and TSTF–284, Revision 3

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

Response: No.

The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses' assumptions. This change is administrative in nature. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Adoption of TSTF–264, Revision 0

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

Response: No.

The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are Consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the justification, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO [limiting condition for operation] can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Adoption of TSTF–299, Revision 0

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

Response: No.

The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change provides additional restrictions which enhance plant safety. This change maintains requirements within the safety analyses and licensing basis. Therefore, this 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: Jonathan Rogoff, Morgan Lewis, 1111 Pennsylvania Avenue NW., Washington, DC 20004.

NRC Section Chief: L. Raghavan.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: February 27, 2004.

Description of amendment request: The proposed amendment would remove license condition 2.C.(2)(b) to perform large transient testing as part of the extended power uprate (EPU) power ascension testing program at the Duane Arnold Energy Center (DAEC).

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

The requested licensing action would remove the current requirement to perform specific large transient tests as part of the DAEC EPU power ascension testing program. No other changes are proposed. Therefore, the probability of an accident previously evaluated is not significantly increased.

The proposed action will not affect any System, Structure, or Component designed for the mitigation of previously analyzed events. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Thus, the proposed change will not increase the consequences of any previously evaluated accident.

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

The requested licensing action would remove the current requirement to perform specific large transient tests as part of the DAEC EPU power ascension testing program. No other changes are proposed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed amendment will not involve a significant reduction in a margin of safety.

Performance of these specific large transient tests is not necessary to ensure acceptable plant operation at the higher thermal power level. Simple, integrated systems tests are performed in lieu of the complex, challenging large transient tests. Other required testing of the specific SSCs that have been modified for EPU ensures that the plant will respond as expected during any abnormal operating event, including these specific transients. Thus, the proposed elimination of the large transient tests will not significantly reduce any margin of safety from that previously approved for EPU operation at the DAEC.

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: Jonathan Rogoff, Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: L. Raghavan.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: January 30, 2004.

Description of amendment request: The proposed amendment would revise Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to (1) clarify the permissive set point for the source range monitor (SRM) detector not-fully-inserted rod block bypass, (2) correct a typographical error in the surveillance requirement for suppression pool temperature monitoring, (3) clarify the set point for the pressure suppression chamber-reactor building vacuum breakers instrumentation, (4) clarify the operating force requirements for the pressure suppression chamber—drywell vacuum breakers surveillance test, and (5) make corrections resulting from License Amendments (LAs) 130 and 132.

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 SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber—Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber—Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. The proposed TS changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. The changes do not affect plant operation, design function or any analysis that verifies the capability of a SSC [structure, system or component] to perform a design function. Further, the proposed changes do not increase the likelihood of the malfunction of any structure, system or component (SSC) or impact any analyzed accident. Consequently, the probability of an accident previously evaluated is not affected.

Therefore, the proposed amendment 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 SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber—Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber—Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. The changes do not create the possibility of new credible failure mechanisms, or malfunctions. These changes do not modify the design function or operation of any SSC. Further the changes do not involve physical alterations of the plant; no new or different type of equipment will be installed. The proposed changes do not introduce new accident initiators. Consequently, the changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber—Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber—Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. These changes do not exceed or alter a design basis or a safety limit for a parameter established in the MNGP Updated Safety Analysis Report (USAR) or the MNGP facility license. Consequently, the changes do not result in a significant reduction in the margin of safety.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves 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.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: February 10, 2004.

Description of amendment request: The proposed change involves the extension from 1 hour to 24 hours of the completion time (CT) for Action (a) of Technical Specification (TS) 3.5.1.1, which defines requirements for accumulators. Accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. Action (a) of TS 3.5.1.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved topical report WCAP–15049–A, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee included in its application several minor changes to make the plant specific TS more consistent with the STS and TSTF–370. The licensee affirmed the applicability of the following NSHC determination in its application dated February 10, 2004.

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

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

The basis for the accumulator limiting condition for operation (LCO), as discussed in Basis Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049–A is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049–A for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,'' and 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049–A, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049–A evaluation as a worst case data point. In addition, WCAP–15049–A states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049–A evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049–A evaluation demonstrates that the small increase in risk due to increasing the CT for an inoperable accumulator is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

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

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Basis Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk.

Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049–A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049–A evaluation.

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: February 20, 2004.

Description of amendment request: The proposed amendments would revise Vogtle Electric Generating Plant, Units 1 and 2 Administrative Controls Section 5.2.2.g of Technical Specification to limit the requirement of the Shift Technical Advisor function to Modes 1–4 in accordance with NUREG 0737.

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 to TS [Technical Specification] 5.2.2.g does not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. This revision does not have any effect on the probability of any accident initiators. The consequences of accidents previously evaluated in the FSAR are not adversely affected by this proposed change because the STA [Shift Technical Advisor] is not credited for mitigation of any accidents. The proposed change which requires the STA function to be available while in Modes 1–4 is in accordance with the requirements of NUREG 0737, Item I.A.1.1. Consequently, the probability or consequences of an accident previously evaluated are not significantly increased.

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

The proposed change to TS 5.2.2.g does not create the possibility of a new or different kind of accident from any previously evaluated. No new accident scenarios, failure mechanism, or limiting single failures are introduced as a result of the proposed change. The proposed Technical Specifications change does not challenge the performance or integrity of any safety-related systems. The proposed change to TS 5.2.2.g is in accordance with NUREG 0737.

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

The proposed change to TS 5.2.2.g will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. The STA function is to evaluate plant conditions and provide advice to the shift supervisor during plant transients and accidents. The proposed change limits the requirements for the STA function to Modes 1–4 in accordance with NUREG 0737. The STA function is not credited for the mitigation of any accidents previously evaluated.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 26, 2004.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)'', to reference the NRC-approved methodology for developing Pressure-Temperature limits and Cold Overpressure Protection System setpoints and the methodology used to justify eliminating the reactor vessel closure head/vessel flange requirements. The proposed amendment would also revise TS 3.4.12, "Cold Overpressure Protection System (COPS)'', to change the Reactor Coolant System vent size.

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

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

No. The proposed changes to the Technical Specifications [TS] and PTLRs [Pressure and Temperature Limits Reports] do not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated by the SSCs [systems, structures and components] as previously analyzed.

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

No. The proposed changes do not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. The changes to the P-T [pressure-temperature] limits and COPS [Cold Overpressure Protection Systems] setpoints will ensure that appropriate fracture toughness margins are maintained to protect against reactor vessel failure during both normal and low temperature operation. The changes to the P-T limits and COPS setpoints are consistent with the methodology approved by the NRC [Nuclear Regulatory Commission] in WCAP–14040, Rev. 4. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses.

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

No. The proposed changes will not adversely affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The utilization of ASME [American Society of Mechanical Engineers] Code Case N–640 maintains the relative margin of safety commensurate with that which existed at the time that ASME B&PV [Boiler and Pressure Vessel] Code, Section XI, Appendix G was approved in 1974 and will ensure an acceptable margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant (BFN), Unit 1, Limestone County, Alabama

Date of amendments request: March 9, 2004 (TS 434).

Description of amendments request: The proposed amendment would lower the current Reactor Vessel Water Level—Low, Level 3 Allowable Value in the Unit 1 Technical Specifications for several instrument functions to reduce the likelihood of unnecessary reactor scrams and the resultant engineered safety feature actuations by increasing the operating range between the normal reactor vessel water level and Level 3 trip functions.

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

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

No. The Reactor Vessel Water Level—Low, Level 3 functions are in response to water level transients and are not involved in the initiation of accidents or transients. Therefore, reducing the BFN, Unit 1, Level 3 Allowable Value does not increase the probability of an accident previously evaluated.

Additionally, the results of the safety evaluation associated with the lowering of the Level 3 Allowable Value concludes that the previously evaluated transient and accident consequences are not significantly affected by the change. Therefore, the proposed amendment does not involve a significant increase in the probability of consequences or an accident previously evaluated.

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

No. The proposed amendment to lower the BFN, Unit 1, Reactor Vessel Water Level—Low, Level 3 Allowable Value does not involve a hardware change and the purpose of the Level 3 function is not affected. The Level 3 functions will continue to fulfill their design objective. The proposed changes do not create the possibility of any new failure mechanisms. No new external threats or release pathways are created. Therefore, reduction of the Allowable Value does not result in the possibility of a new or different kind of accident.

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

No. The results of the safety evaluation associated with the reducing the BFN, Unit 1, Reactor Vessel Water Level—Low, Level 3 Allowable Value concluded that transient and accident consequences remain within the required acceptance criteria. Therefore, the margin of safety is not reduced for any event evaluated.

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

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

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: March 5, 2004.

Description of amendment request: The proposed amendments would delete Technical Specifications (TSs) 3.6.4.1, "Hydrogen Monitors,'' and 3.6.4.2, "Electric Hydrogen Recombiners-W.'' The proposed changes support Title 10, Code of Federal Regulations, Part 50, Section 44 (10 CFR 50.44), "Standards for Combustible Gas Control system in Light-Water-Cooled Power Reactors'' and are consistent with the Industry/Technical Specification Task Force (TSTF) Standard TS Change Traveler, TSTF–447, "Elimination of Hydrogen Recombiners and change to Hydrogen and Oxygen Monitors.''

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:

TVA has reviewed the proposed no significant hazards consideration determination published on September 25, 2003, (68 FR 55416) as part of the consolidated line item improvement process (CLIIP). TVA has concluded that the proposed determination presented in the notice is applicable to SQN, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

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

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

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: March 5, 2004.

Description of amendment request: The proposed amendments would delete surveillance requirement (SR) 4.9.2.c and SRs 4.10.3.2 and 4.10.4.2 from the Technical Specifications (TSs). SR 4.9.2.c requires channel functional tests for each Source Range neutron flux monitor within 8 hours prior to initial core alterations. SRs 4.10.3.2 and 4.10.4.2 require channel functional tests for each Power Range and Intermediate Range neutron flux monitor within 12 hours prior to the initiation of a physics test. In addition, the proposed changes include revisions to the associated TS bases (3/4.9.2, 3/4.10.3, and 3/4.10.4).

Basis for proposed no significant hazards consideration determination: As required by Title 10, Code of Federal Regulations, Part 50, Section 91(a) (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 amendment removes the requirement to perform an additional CHANNEL FUNCTIONAL TEST (CFT) on the Intermediate and Power Range functions within 12 hours of performing a PHYSICS TEST. The Intermediate and Power Range instrumentation is determined to be OPERABLE by periodic SRs which must be confirmed to be within frequency prior to making the reactor critical. The proposed amendment also removes the requirement to perform an additional CFT on the Source Range monitors. The Source Range instrumentation is determined to be OPERABLE by periodic SRs, which must be confirmed to be within frequency prior to Mode 6, prior to CORE ALTERATIONS, and must remain OPERABLE. A CFT for the Source Range, Intermediate Range, or Power Range instrumentation is not a precursor to, or assumed to be an initiator of any analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated.

Regarding a significant increase in the consequences of an accident, several factors must be considered. First the PHYSICS TESTS are performed in accordance with the TSs in Mode 2. Therefore, the power level of the reactor is limited to 5 percent or less. Along with this, the reactor trip function of the Intermediate Range detectors will be unaffected by the proposed amendment and therefore, will be available to mitigate a reactivity transient at low power. Further, the trip setpoint for the Power Range monitors are decreased during startup. This setpoint reduction provides an additional measure to limit a reactivity excursion. 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?

No. The proposed changes permit the conduct of normal operating evolutions during limited periods when additional controls over reactivity margin are imposed by the TSs. The proposed change does not introduce any new equipment into the plant or significantly alter the manner in which existing equipment will be operated. The proposed changes are not based on a change in the design or configuration of the plant. The changes to operating allowances are minor and are only applicable during certain conditions. The operating allowances are consistent with those acceptable at other times. The proposed changes delete the requirements for the performance of a CFT for the Source Range, Intermediate Range, and Power Range instrumentation within 8 hours of initiating CORE ALTERATIONS for the Source Range monitors and within 12 hours of starting a PHYSICS TEST for the Intermediate Range and Power Range instrumentation. Since the proposed changes only allow activities that are presently approved and routinely conducted, no possibility exists for a new or different kind of accident from those previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. As stated previously, the proposed change deletes the requirement to perform an additional CFT for the Source Range, Intermediate Range, and Power Range instrumentation within 8 hours of initiating CORE ALTERATIONS for the Source Range monitors and within 12 hours of starting a PHYSICS TEST for the Intermediate Range and Power Range instrumentation. The Source Range, Intermediate Range, and Power Range instrumentation channels are determined to be OPERABLE by meeting the requirements of the periodic surveillance. These SRs are not affected by the proposed amendment. The proposed changes do not involve a significant reduction in a margin of safety because the ability to monitor the reactor during the applicable operating conditions and modes of operation will be maintained. The proposed changes do not affect these operating restrictions and the margin of safety which assures the ability to monitor the reactor is not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: March 5, 2004.

Description of amendment request: The proposed amendments would change Technical Specification (TS) 4.0.5.c. Specifically, the proposed change would extend the examination frequency for the reactor coolant pump (RCP) motor flywheel from a 10-year interval to an interval not to exceed 20 years. This proposed change is consistent with the Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–421, "Revision to RCP Flywheel Inspection Program (WCAP–15666).''

Basis for proposed no significant hazards consideration determination: As required by Title 10, *Code of Federal Regulations,* Part 50, Section 91(a) (10 CFR 50.91(a)), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has reviewed the proposed no significant hazards consideration determination published on June 24, 2003 (68 FR 37590), as part of the consolidated line item improvement process (CLIIP). TVA has concluded that the proposed determination presented in the notice is applicable to SQN, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

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

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

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: March 4, 2004.

Description of amendment request: The proposed amendments would delete the note in Improved Technical Specification Surveillance Requirement 3.4.12.7 that permitted the performance of the Channel Operational Test within 12 hours of entering a mode in which the power-operated relief valves (PORVs) are required to be operable for low temperature overpressure protection (LTOP).

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 to perform a Channel Operational Test on each required PORV at least 31 days prior to entering the LTOP Mode will continue to ensure verification and adjustment, if required, of its lift setpoint. Changes will not affect the probability of occurrence of any accident previously analyzed: nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. Therefore, the proposed changes do not involve a significant increase in the consequences of any previously analyzed accident.

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

The proposed changes to perform a Channel Operational Test on each required PORV at least 31 days prior to entering the LTOP Mode will not create any new accident or event initiators. No systems, structures, or components are being physically modified such that the design function is being altered. The proposed changes do not impose any new or different requirements for the performance of the Channel Operational Test. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from those previously analyzed.

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

The proposed changes do not involve any change to the safety analysis limits. The level of safety of facility operation is unaffected by the proposed changes since there is no change in the intent for the performance of the Channel Operational Test. Therefore, it is concluded that the margin of safety will not be reduced by the implementation of the changes.

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.

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

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

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

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

Date of application for amendment: February 25, 2004.

Brief description of amendments: The amendment would extend the implementation date for Amendment Nos. 261 and 238 for Calvert Cliffs Units 1 and 2, respectively, to July 1, 2004. The changes to the reactor pressure vessel pressure-temperature limits cooldown rates that were approved by Amendment Nos. 261 and 238 are more conservative than the plants existing rates and result in a longer cooldown period. The existing cooldown rates are acceptable through the end of 2004.

Date of publication of individual notice in ***Federal Register:*** March 5, 2004 (69 FR 10487).

Expiration date of individual notice: May 5, 2004.

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, 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 NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: December 23, 2003, as supplemented by letter dated January 30, 2004.

Brief description of amendment: The amendment modified Technical Specification (TS) requirements for mode change limitations to adopt the TS Task Force (TSTF) change TSTF–359, "Increase Flexibility in Mode Restraints.''

Date of issuance: March 29, 2004.

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

Amendment No.: 241.

Facility Operating License No. NPF–69: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 20, 2004 (69 FR 2738).

The January 30, 2004, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The staff's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 7, 2003, and its supplement dated December 18, 2003.

Brief description of amendments: The amendments revise Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The amendments also delete the provisions of Surveillance Requirement (SR) 3.0.2 from this TS. In addition, the amendments revise TS 5.5.16, "Containment Leakage Rate Testing Program,'' to add exceptions to Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program.'' Also, the paragraphs in Section 5.5.16 have been sequenced to more clearly separate the requirements of the program. This is considered an administrative change and is consistent with the guidance in NUREG–1432, "Standard Technical Specifications Combustion Engineering Plants,'' Revision 2.

Date of issuance: March 19, 2004.

Effective date: March 19, 2004, and shall be implemented within 90 days of the date of issuance.

Amendment Nos.: Unit 1–151, Unit 2—151, Unit 3—151.

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

*Date of initial notice in **Federal Register:*** December 9, 2003 (68 FR 68659) The December 18, 2003, supplemental letter provided revised technical specification pages to reflect changes that were approved in Amendment No. 149, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 19, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 17, 2003 as supplemented by letter dated February 20, 2004.

Brief description of amendments: The amendments revise the technical specifications to support the replacement of part-length control element assemblies (CEAs) with a new design, referred to as part-strength CEAs. The two designs are geometrically very similar and contain essentially the same amount and type of neutron absorber in the lower half of the assemblies, which is the region of the CEAs inserted into the reactor core during normal operations.

Date of issuance: March 23, 2004.

Effective date: March 23, 2004, and shall be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1—152, Unit 2—152, Unit 3—152.

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

*Date of initial notice in **Federal Register:*** December 9, 2003 (68 FR 68657). The February 20, 2004, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 23, 2004.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of application for amendment: October 31, 2003, as supplemented March 4, March 12, and March 19, 2004.

Brief description of amendment: The amendment revised the Minimum Critical Power Ratio Safety Limit contained in Technical Specification 2.1.1.2.

Date of issuance: March 26, 2004.

Effective date: Effective as of the date of issuance and shall be implemented prior to startup for Unit 1, Cycle 15, operation.

Amendment No.: 231.

Facility Operating License Nos. DPR–71: Amendment changes the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 6, 2004 (69 FR 693). The March 4, March 12, and March 19, 2004, supplemental letters provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register** and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 14, 2003, as supplemented by letters dated November 10 and December 10, 2003, and January 30, 2004.

Brief description of amendment: This amendment revises Technical Specification (TS) 5.6.3.d to allow an increase in the decay heat load from 1.0 MBTU/hr to 7.0 MBTU/hr for fuel stored in Spent Fuel Pools C and D at Shearon Harris Nuclear Power Plant, Unit 1.

Date of issuance: March 26, 2004.

Effective date: March 26, 2004.

Amendment No.: 115.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 18, 2003 (68 FR 12948). The November 10 and December 10, 2003, and January 30, 2004, supplements provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register** and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 2004.

No significant hazards consideration comments received: No.

Consumers Energy Company, Docket No. 50–155, Big Rock Point Nuclear Plant, Charlevoix County, Michigan

Date of application for amendment: November 20, 2002, and August 6, 2003, as supplemented by letters dated December 1, 2003, and February 20, 2004.

Brief description of amendment: The amendment revises the Big Rock Point License and Defueled Technical Specifications to remove reactor operational and administrative requirements that are no longer applicable due to the transfer of all spent fuel from the spent fuel pool into dry cask storage at the Big Rock Point Independent Spent Fuel Storage Installation.

Date of issuance: March 19, 2004.

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

Amendment No.: 125.

Facility Operating License No. DPR–6: Amendment revises the Defueled Technical Specifications.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2800), and November 25, 2003 (68 FR 66133). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 19, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 4, 2003, as supplemented May 13 and September 18, 2003, and February 12 and March 10, 2004.

Brief description of amendment: The amendment revised selected sections of the Technical Specifications (TSs) based upon a re-analysis of fuel handling accidents (FHAs). The revised analysis is based upon selective implementation of the alternative source term methodology of Regulatory Guide 1.183, and in accordance with Title 10 of the Code of Federal Regulations, Section 50.67. Specifically, the amendment revised: TS 3.7.8, "Plant Systems, Control Room Envelope Pressurization System;'' TS 3.9.4, "Refueling Operations, Containment Building Penetrations;'' TS 3.9.9, "Refueling Operations, Containment Purge and Exhaust Isolation System,'' and TS 3.9.12, "Refueling Operations, Fuel Building Exhaust Filter System.''

Date of issuance: March 17, 2004.

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

Amendment No.: 219.

Facility Operating License No. NPF–49: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** March 4, 2003 (68 FR 40711). The May 13 and September 18, 2003, and February 12 and March 10, 2004, supplements contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2004.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: December 5, 2003, as supplemented on February 9, 2004.

Brief description of amendment: The amendment revised the Safety Limit Minimum Critical Power Ratio values in Technical Specification 1.1.A.1 to incorporate the results of the cycle-specific core reload analysis for Vermont Yankee Nuclear Power Station Cycle 24 operation.

Date of Issuance: March 22, 2004.

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

Amendment No.: 217.

Facility Operating License No. DPR–28: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** January 20, 2004 (69 FR 2741). The supplement dated February 9, 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.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 22, 2004.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: March 26, 2003, as supplemented on July 24, 2003.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) regarding reactor pressure vessel (RPV) fracture toughness and material surveillance requirements (SRs). Specifically, the amendment revised the pressure-temperature limits for the RPV as specified in TS Figures 3.6.1, 3.6.2, and 3.6.3. In addition, the amendment deleted TS 4.6.A.5, which specifies plant-specific RPV material SRs. These plant-specific SRs are being replaced by implementing the Boiling Water Reactor Vessel and Internals Project (BWRVIP) RPV integrated surveillance program (ISP). The details of the BWRVIP ISP will be added to the Vermont Yankee Nuclear Power Station Updated Final Safety Analysis Report.

Date of Issuance: March 29, 2004.

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

Amendment No.: 218.

Facility Operating License No. DPR–28: Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** April 29, 2003 (68 FR 22747). The supplement dated July 24, 2003, 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 this amendment is contained in a Safety Evaluation dated March 29, 2004.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 11, 2003, as supplemented on December 5, December 30, 2003, and February 18, 2004.

Brief description of amendments: The amendments revise technical specification 3.7.8 to permit a one-time extension from 72 hours to 144 hours for the completion time required to restore a unit specific essential service water train to operable status.

Date of issuance: March 18, 2004.

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

Amendment Nos.: 136/136, 130/130.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 30, 2003.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 18, 2004.

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: September 8, 2003.

Brief description of amendments: The amendments modified Technical Specifications requirements to adopt the provisions of Industry/Technical Specification Task Force (TSTF) change 359, "Increase Flexibility in Mode Restraints.''

Date of issuance: March 12, 2004.

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

Amendment Nos.: 169 and 132.

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

*Date of initial notice in **Federal Register:*** December 9, 2003 (68 FR 68668).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 12, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 15, 2002, as supplemented by letter dated January 14, 2004.

Description of amendment request: The amendment revises the Technical Specifications (TSs) to relocate the boron concentration limits and "Safety Limits'' figures to the Core Operating Limits Report. Some limiting conditions and actions are revised to be consistent with the Improved Standard Technical Specifications.

Date of issuance: March 23, 2004.

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

Amendment No.: 96.

Facility Operating License No. NPF–86: The amendment revises the TS.

*Date of initial notice in **Federal Register:*** May 28, 2002 (67 FR 36931). The January 14, 2004, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 23, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 25, 2003.

Brief description of amendment: The amendment revises the Technical Specification (TS) for Limiting Condition for Operation requirement 3.5.1 to incorporate TS Task Force Traveler 318 to allow one low pressure coolant injection pump inoperable in each of the two emergency core cooling system divisions.

Date of issuance: March 31, 2004.

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

Amendment No.: 203.

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

*Date of initial notice in **Federal Register:*** October 14, 2003 (68 FR 59218).

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: March 27, 2003, as supplemented on November 3, 2003, and January 28, 2004.

Brief description of amendments: The amendment revises Technical Specification Surveillance Requirement 3.2.4.2, "Rod Group Alignment Limits.'' The revision expands the alignment limits on allowable rod cluster control assembly, or rod, deviation from demanded position. The change applies in Mode 1, when operating at greater than 85 percent of rated thermal power.

Date of issuance: March 29, 2004.

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

Amendment Nos.: 212 and 217.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 29, 2003 (68 FR 22749).

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 29, 2004.

No significant hazards consideration comments received: No.

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 application for amendments: June 11, 2003.

Brief description of amendments: The amendments revise the technical specifications to allow use of the power distribution monitoring system (PDMS) for power distribution measurements as described in Topical Report WCAP–12462–P–A, "BEACON: Core Monitoring and Support System.''

Date of issuance: March 31, 2004.

Effective date: March 31, 2004, and shall be implemented within 180 days from the date of issuance.

Amendment Nos.: Unit 1—164; Unit 2—166.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 8, 2003 (68 FR 40717).

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

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 2 and 3, Limestone County, Alabama

Date of application for amendments: September 18, 2003, as supplemented December 8, 2003, and February 24, 2004.

Description of amendment request: The amendments revised the pressure-temperature limit curves in Technical Specification (TS) 3.4.9.

Date of issuance: March 10, 2004.

Effective date: March 10, 2004.

Amendment Nos.: 288 & 247.

Facility Operating License No. DPR–52 and DPR–68: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** October 28, 2003 (68 FR 61480). The December 8, 2003, and February 24, 2004, letters provided clarifying information that did not change the scope of the original request or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 10, 2004.

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: March 24, 2003, as supplemented December 4, 2003, and February 12, 2004.

Brief description of amendment: The amendment revises the design and licensing basis failure modes and effects analysis for specific valves in the essential raw cooling water system, component cooling water system, and control air system to address a condition in which containment integrity, accident flood levels, and sump boron concentrations subsequent to a high-energy line break could not be automatically ensured, and, therefore, manual actions are required.

Date of issuance: March 29, 2004.

Effective date: As of the date of issuance and shall be implemented in conjunction with the next update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e).

Amendment No.: 51.

Facility Operating License No. NPF–90: Amendment revises the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18287). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 29, 2004.

No significant hazards consideration comments received: No.

TXU Generation Company LP, Docket No. 50–445, Comanche Peak Steam Electric Station, Unit No. 1, Somervell County, Texas

Date of amendment request: July 21, 2003, as supplemented by letters dated January 8, January 21, and March 8, 2004.

Brief description of amendments: The Amendment revises the Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program,'' to allow the use of Westinghouse (Westinghouse Electric Station LLC) leak limiting Alloy 800 sleeves for repair of degraded SG tubes.

Date of issuance: March 24, 2004.

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

Amendment Nos.: 112.

Facility Operating License No. NPF–87: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 21, 2003. Supplemental letters dated January 8, January 21, and March 8, 2004 provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 24, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 8, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The amendment also deletes the provisions of Surveillance Requirement (SR) 3.0.2 from this TS. In addition, the amendment revises TS 5.5.16, "Containment Leakage Rate Testing Program,'' to add exceptions to Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program.''

Date of issuance: March 17, 2004.

Effective date: March 17, 2004, and shall be implemented within 90 days from the date of issuance.

Amendment No.: 160.

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

Date of initial notice in ***Federal Register:*** January 6, 2004 (69 FR 700).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2004.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: October 17, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The amendment also deletes the provisions of Surveillance Requirement (SR) 3.0.2 from this TS. In addition, the amendment revises TS 5.5.16, "Containment Leakage Rate Testing Program,'' to add exceptions to Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program.''

Date of issuance: March 17, 2004.

Effective date: March 17, 2004, and shall be implemented within 90 days from the date of issuance.

Amendment No.: 152.

Facility Operating License No. NPF–42: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2003 (68 FR 64140).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 5th day of April 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–8047 Filed 4–12–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission (NRC) has issued errata sheets for two guides in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in its review of applications for permits and licenses, and data needed by the NRC staff in its review of applications for permits and licenses.

Errata sheets have been issued for Regulatory Guide 1.184, "Decommissioning of Nuclear Power Reactors,'' and Regulatory Guide 1.185, "Standard Format and Content for Post-Shutdown Decommissioning Activities Report.'' These errata sheets update Reference 1 in both guides to Supplement 1, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities'' (Volumes 1 and 2) to NUREG–0586 (November 2002), which supersedes the previous version of NUREG–0586, issued in August 1988.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Questions on the content of this guide may be directed to Mr. T. Smith, (301) 415–6721; e-mail *tbs1@nrc.gov.*

Regulatory guides are available for inspection or downloading at the NRC's Web site at *http://www.nrc.gov* under NRC Documents and in NRC's ADAMS System at the same site. Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301) 415–2289, or by e-mail to *distribution@nrc.gov.* Issued guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; telephone 1–800–553–6847; *http://www.ntis.gov/.* Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

—(5 U.S.C. 552(a))

Dated at Rockville, MD, this 31st day of March 2004.

For the Nuclear Regulatory Commission.

Ashok C. Thadani,

Director, Office of Nuclear Regulatory Research.

[FR Doc. 04–8287 Filed 4–12–04; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[File No. 1–31703]

Issuer Delisting; Notice of Application of Essex Corporation, To Withdraw Its Common Stock, No Par Value, From Listing and Registration on the American Stock Exchange LLC

April 7, 2004.

Essex Corporation, a Virginia corporation ("Issuer''), has filed an application with the Securities and Exchange Commission ("Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 ("Act'') [1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Common Stock, no par value ("Security''), from listing and registration on the American Stock Exchange LLC ("Amex'' or "Exchange'').

The Board of Directors ("Board'') of the Issuer approved a resolution on March 15, 2004 to withdraw the Issuer's Security from listing on the Amex and to list the Security on Nasdaq National Market System ("Nasdaq NMS''). The Board states that the reasons it is taking such action are to offer shareholders a broader market, including liquidity and increased visibility. The Issuer expects to trade the Security on the Nasdaq NMS on March 31, 2004.

The Issuer stated in its application that it has met the requirements of Amex Rule l8 by complying with all applicable laws in the State of Virginia, in which it is incorporated, and with the Amex's rules governing an issuer's voluntary withdrawal of a security from listing and registration.

The Issuer's application relates solely to the withdrawal of the Securities from listing on the Amex and from registration under section 12(b) of the Act [3] and shall not affect its obligation to be registered under section 12(g) of the Act.[4] Any interested person may, on or before April 30, 2004, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549–0609, facts bearing upon whether the application has been made in accordance with the rules of the Amex and what terms, if any, should be imposed by the Commission for the protection of investors. All comment letters should refer to File No. 1–31703. Comments may also be submitted electronically at the following e-mail address: *rule-comments@sec.gov.* The

[1] 15 U.S.C. 78l(d).

[2] 17 CFR 240.12d2–2(d).

[3] 15 U.S.C. 781(b).

[4] 15 U.S.C. 781(g).