NUCLEAR REGULATORY COMMISSION

Meetings; Sunshine Act

DATE: Weeks of January 31, February 7, 14, 21, 28, March 7, 2005.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of January 31, 2005

Thursday, February 3, 2005

9:30 a.m. Briefing on Human Capital Initiatives (Closed—Ex. 2).

Week of February 7, 2005—Tentative

There are no meetings scheduled for the week of February 7, 2005.

Week of February 14, 2005—Tentative

Tuesday, February 15, 2005

9:30 a.m. Briefing on Office of Nuclear Material Safety and Safeguards Programs, Performance, and Plans—Waste Safety (Public Meeting) (Contact: Jessica Shin, 301–415–8117).

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

Week of February 21, 2005—Tentative

Tuesday, February 22, 2005

9:30 a.m. Briefing on Status of Office of the Chief Information Officer (OCIO) Programs, Performance, and Plans (Public Meeting) (Contact: Patricia Wolfe, 301–415–6031).

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

1:30 p.m. Briefing on Emergency Preparedness Program Initiatives (Closed—Ex. 1) (This meeting was originally scheduled for February 15, 2005).

Wednesday, February 23, 2005

9:30 a.m. Briefing on Status of Office of the Chief Financial Officer (OCFO) Programs, Performance, and Plans (Public Meeting) (Contact: Edward New, 301–415–5646).

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

Thursday, February 24, 2005

1 p.m. Briefing on Nuclear Fuel Performance (Public Meeting) (Contact: Frank Akstulewicz, 301–415–1136).

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

Week of February 28, 2005—Tentative

There are no meetings scheduled for the week of February 28, 2005.

Week of March 7, 2005—Tentative

Monday, March 7, 2005

9:30 a.m. Briefing on Office of Nuclear Material Safety and Safeguards Programs, Performance, and Plans—Materials Safety (Public Meeting) (Contact: Shamica Walker, 301–415–5142).

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

* The schedule 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.

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

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–415–2100, or by e-mail at *aks.@nrc.gov*. Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

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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: January 27, 2005.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 05–1885 Filed 1–28–05; 9:39 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 January 7, 2005, through January 19, 2005. The last biweekly notice was published on January 18, 2005 (70 FR 2886).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Date of amendment request: September 15, 2004.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain hydrogen recombiners and hydrogen and oxygen monitors. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on September 25, 2003 (68 FR 55416). Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by order for many facilities and were added to, or included, in the TSs for nuclear power reactors currently licensed to operate. The revised Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, ''Standards for combustible gas control system in light-water-cooled power reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated September 15, 2004.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 NRC 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 and oxygen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen and oxygen 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 NRC 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. Also, as part of the rulemaking to revise 10 CFR 50.44, the NRC found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen 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, [classification of the oxygen monitors as Category 2,] and removal of the hydrogen and oxygen monitors from TS will not prevent an accident management strategy through the use of the severe accident management guidelines, the emergency plan, the emergency operating procedures, and site survey monitoring that support modification of emergency plan protective action recommendations.

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen 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 and oxygen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen and oxygen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen and oxygen 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 and oxygen 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 NRC 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.

Category 2 oxygen monitors are adequate to verify the status of an inerted containment.

Therefore, this change does not involve a significant reduction in the margin of safety. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors. Removal of hydrogen and oxygen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

The NRC staff proposes to determine that the amendment request involves NSHC.

Attorney for licensee: Mr. Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60666.

NRC Section Chief: Gene Y. Suh.

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

Date of amendments request: December 16, 2004.

Description of amendments request: The requested change will delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 16, 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:

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating letter report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance hazards consideration.

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Section Chief: Robert A. Gramm.

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 1, 2004.

Description of amendments request: The requested change will delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 1, 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:

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance 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.

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

Date of amendment request: December 6, 2004.

Description of amendment request: The requested change will delete

Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 6, 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:

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

Response: No.

The proposed change eliminates the TSs reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significant hazards consideration.

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

NRC Section Chief: M. Kotzalas (Acting).

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

Date of amendment request: June 6, 2004.

Description of amendment request: The proposed change would modify the Millstone Power Station, Unit No. 2 Technical Specifications (TSs) to extend the 10-year test interval for the Integrated Leakage Rate Test program to 15 years from the last Type A test. Specifically, the proposed change would revise TS 6.19, ''Containment Leakage Rate Testing [CLRT] Program,'' and permit a one-time, 5-year extension of the 10-year performance-based Type A test interval. In addition, the testing would be in accordance with the CLRT Program, Regulatory Guide (RG) 1.163, ''Performance-Based Containment Leak-Test Program'' and surveillance testing requirements as proposed in Nuclear Energy Institute 94–01 for Type A testing.

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 license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence, nor is it an activity or modification that by itself could lead to equipment failure or accident initiation.

The proposed one-time, five-year extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that even reducing the Type A (ILRT [integrated leak rate test]) testing frequency to once per twenty years leads to an imperceptible increase in risk.

DNC (the licensee) provides a high degree of assurance through indirect testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] are also performed in order to identify indications of containment degradation that could affect leak-tightness. Separately, Type B and C testing required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that a one-time, five-year extension to the Millstone Unit 2 Type A test interval will not represent a significant increase in the consequences of an accident.

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

The proposed revision to the Technical Specifications adds a one-time extension to the current interval for Type A testing for Millstone Unit 2. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

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

The proposed revision to Millstone Unit 2 Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Millstone Unit 2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF [core damage frequency] below 10^{-6}/yr and increases in LERF [large early release frequency] below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF, resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 0.83×10^{-8}/yr, based on internal events. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from ten to fifteen years is, therefore, considered non-risk significant and will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 generically concludes that the design containment leakage rate contributes about 0.1 percent of the overall risk. Decreasing the Type A testing frequency would have a minimal effect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. Given that the proposed change will continue to meet the current design basis, any reduction in a margin of safety would not be significant.

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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: December 16, 2004.

Description of amendment request: The proposed amendment would revise the current fuel rod average licensing basis burnup limit for one lead test assembly (LTA) containing advanced zirconium based alloys to a limit not exceeding 71,000 MWD/MTU.

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

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

The Westinghouse LTA is very similar in design to the Westinghouse fuel that comprises the remainder of the core. The reload core design for Millstone Unit 3 Cycle 12, where one LTA will operate to high burnup, will meet all applicable design criteria. The performance of the Emergency Core Cooling System will not be affected by the operation of the LTA and operation of the LTA to high burnup will not result in a change to the Millstone Unit 3 reload design and safety analysis limits. Operation of one Westinghouse LTA to high burnup will not result in a measurable impact on normal operating releases, and will not increase the predicted radiological consequences of accidents postulated in Chapter 15 of the Millstone FSAR [final safety analysis report]. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

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

The Westinghouse LTA is very similar in design (both mechanical and composition of materials) to the resident Westinghouse fuel. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The irradiation of one LTA to high burnup does not involve any alteration to plant equipment or procedures, which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

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

The operation of one Westinghouse LTA to high burnup does not change the performance requirements of any system or component such that any design criteria will be exceeded. The normal limits on core operation defined in the Millstone Unit 3 Technical Specifications will remain applicable for the core in which the high burnup assembly is irradiated. Therefore, the margin of safety as defined in the Bases to the Millstone Unit 3 Technical Specifications is not significantly reduced.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: Darrell Roberts.

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

Date of amendment request: September 8, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated September 8, 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 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. Category 1 in RG 1.97 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 severe accident management guidelines (SAMGs), the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TSs, 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 TSs, 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 TSs, 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 TSs will not result in a significant reduction in their functionality, reliability, and availability.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: Darrell J. Roberts.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina; 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: September 20, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated September 20, 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 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. Category 1 in [Regulatory Guide] RG 1.97 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 [Technical Specification] TS will not prevent an accident management strategy through the use of the severe accident management guidelines (SAMGs), the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

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

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

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

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

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

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

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

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

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

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.

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: October 22, 2004.

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

The Nuclear Regulatory Commission (NRC) staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated October 22, 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 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. Category 1 in RG 1.97 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 severe accident management guidelines (SAMGs), the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

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

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

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

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

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

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

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

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

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

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

Date of amendment request: October 25, 2004.

Description of amendment request: The requested change will delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated October 25, 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:

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in [a] margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance hazards consideration.

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: December 30, 2004.

Description of amendment request: The proposed amendment would revise a Technical Specification (TS) surveillance requirement (SR) in TS 3.1.4, ''Control Rod Scram Times.'' Specifically, the proposed change would revise the frequency for SR 3.1.4.2, ''Control Rod Scram Time Testing,'' from ''120 days cumulative operation in MODE 1'' to ''200 days cumulative operation in MODE 1.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in licensing amendment applications in the **Federal Register** on August 23, 2004 (69 FR 51864). The licensee affirmed the applicability of the model NSHC determination in its application dated December 30, 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:

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The frequency of surveillance testing is not an initiator of any accident previously evaluated. The frequency of surveillance testing does not affect the ability to mitigate any accident previously evaluated, as the tested component is still required to be operable. 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 extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change does not result in any new or different modes of plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change continues to test the control rod scram time to ensure the assumptions in the safety analysis are protected. 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 Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: December 20, 2004.

Description of amendment request: The proposed amendment would increase the lifting tripod's rating from 150 tons to 190 tons. This would allow for additional flexibility when lifting the new reactor vessel head during refueling outages.

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 ANO–1 [Arkansas Nuclear One, Unit 1] Tripod does not perform a safety function required by 10 CFR [Part] 50. The Tripod serves to perform heavy load movements during refueling outages[,] including [movement of] the reactor vessel head. Safe load paths have been established in accordance with NUREG–0612[, ''Control of Heavy Loads at Nuclear Power Plants,''] to ensure that the fuel and safety[-]related equipment required to be inservice are protected. Use of actual Tripod eyelet Certified Material Test Reports (CMTRs) demonstrates that a safety factor of 3 to yield is maintained and that the lifting devices will perform their design function under maximum lifted loads. The Tripod does not serve any mitigative functions to lessen accidents.

Therefore, the proposed change does not affect the probability or consequences of any ANO–1 analyzed accidents.

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

Response: No.

The only time that the Tripod is performing heavy loads movements is during Refueling operations. Safe load paths and load drop analyses have been performed to assure that heavy loads movements will not cause fuel damage or cause safety[-]related equipment to become inoperable. The proposed use of CMTRs instead of minimum yield strength of the material still assures that the Tripod will perform its required function to not create an accident. In addition, there is no change to the operation of the Tripod that would create a new failure mode or possible accident.

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

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

Response: No.

The design margin for the Tripod is established by NUREG–0612 and ANSI [American National Standards Institute] N14.6–1978[, ''Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials'']. A factor of safety of 3 for yield strength and 5 for ultimate strength for both the static and dynamic load factors is required to be met. These factors of safety provide sufficient margin to assure that the Tripod will perform its design function of maximum lifted loads. In addition, the use [of] a dynamic load factor of 1.15 above the static load is well above the actual dynamic factor to be experienced from the design lift speed of the polar crane. The use of CMTRs does not result in a significant reduction in the margin of safety of the Tripod. In addition, the Tripod will be load tested to 150% [percent] of its design static and dynamic loading which will further assure adequate safety margin.

Therefore, the margin of safety is not changed by the proposed change to the ANO–1 SAR [Safety Analysis Report].

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: December 17, 2004.

Description of amendment request: The proposed change will revise the air lock surveillance test acceptance criteria to be consistent with the NRC approved Industry Technical Specification Task Force (TSTF) change to the Standard Technical Specifications (STS), TSTF–52, entitled ''Implement 10 CFR [Part] 50, Appendix J, Option B.'' By letter dated April 6, 1998, the NRC Staff issued amendment number 135 to the GGNS license permitting the implementation of the containment leak rate testing provisions of 10 CFR Part 50, Appendix J, Option B.

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.

Primary containment air lock leak rate testing can have no effect on the probability of any postulated accident. The proposed change will increase the allowed containment air lock leakage rate and convert it from an absolute leakage rate to a percentage of the overall primary containment leakage rate. No change to the overall leakage rate of the containment is being proposed, therefore there is no change to the consequences of any postulated accident. The change in air lock leakage rate will not impact the design or operation of any plant system or component nor will they affect initiation or mitigation of any accidents previously analyzed.

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 primary containment air locks form part of the primary containment pressure boundary. The periodic containment air lock leakage rate tests specified in SR 3.6.1.2.1 verifies that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. This request involves a change in the allowable leakage rate of the primary containment air locks without increasing the overall allowed leakage rate of the containment. Changing the allowable leakage rate has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. There will be no effect on the types and amounts of overall leakage from the primary containment boundary. The proposed amendment will not produce any changes to the design or operation of the plant. The method of performing the test is not changed. No new accident modes are created by changing the allowable leakage in this manner. No safety-related equipment or safety functions are altered as a result of this change.

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.

Air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a design basis accident. The periodic containment air lock leakage rate tests verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. Since no changes are proposed to the maximum allowable primary containment leakage rate, the design basis radiological analysis is not impacted by this change. The license amendment request removes unnecessary conservatism from the testing program and allows consistency with current industry practice. Since no changes are proposed to the maximum allowable primary containment leakage rate, the design basis radiological analysis is not impacted by this change.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Michael K. Webb.

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; Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: September 15, 2004.

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

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated September 15, 2004.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 and oxygen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen and oxygen 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. Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen 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, classification of the oxygen monitors as Category 2, and removal of the hydrogen and oxygen 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 (EOPs), 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 and oxygen 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 and oxygen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen and oxygen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen and oxygen 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 and oxygen 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.

Category 2 oxygen monitors are adequate to verify the status of an inerted containment.

Therefore, this change does not involve a significant reduction in the margin of safety. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors. Removal of hydrogen and oxygen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

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

NRC Section Chief: Gene Y. Suh.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station (DNPS), Units 2 and 3, Grundy County, Illinois; Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, Rock Island County, Illinois

Date of amendment request: November 4, 2004.

Description of amendment request: The proposed amendments would revise the plant technical specification (TS) pressure and temperature (P/T) limit curves for 54 effective full power years (EFPY) to support a 20-year license extension for both DNPS and QCNPS to 60 years (*i.e.*, 54 EFPY), and resolves a non-conservative condition for TS Section 3.4.9, Figure 3.4.9–2, ''Non-Nuclear Heatup/Cooldown Curve,'' for QCNPS.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) section 50.91(a), Exelon Generation Company (EGC) has provided its analysis of the issue of no significant hazards consideration (NSHC), which is presented below:

According to 10 CFR 50.92, ''Issuance of amendment,'' paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if 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.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

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

The proposed changes request that, for DNPS, Units 2 and 3 and QCNPS, Units 1 and 2, P/T limit curves in TS 3.4.9, ''RCS Pressure and Temperature (P/T) Limits,'' be revised.

The P/T limits are prescribed during all operational conditions to avoid encountering pressure, temperature, and temperature rate-of-change conditions that might cause undetected flaws to propagate, resulting in non-ductile failure of the reactor coolant pressure boundary, which is an unanalyzed condition. The methodology used to determine the P/T limits has been approved by the NRC [Nuclear Regulatory Commission] and thus is an acceptable method for determining these limits. Therefore, the proposed changes do not affect the probability of an accident previously evaluated.

There is no specific accident that postulates a non-ductile failure of the reactor coolant pressure (RCP) boundary. The loss of coolant accident analyzed for the plant assumes a 4.281 square feet complete break of the recirculation pump suction line. The revision to the P/T limits does not change this assumption. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes do not change the response of plant equipment to transient conditions. The proposed changes do not introduce any new equipment, modes of system operation, or failure mechanisms.

Non-ductile failure of the RCP boundary is not an analyzed accident. The proposed changes to the P/T limits were developed using an NRC-approved methodology, and thus the revised limits will continue to provide protection against non-ductile failure of the RCP boundary.

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

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

The margin of safety related to the proposed changes is the margin between the proposed P/T limits and the pressures and temperatures that would produce nonductile failure of the RCP boundary. NRC requirements to protect the integrity of the reactor coolant pressure boundary in nuclear power plants is established in 10 CFR 50, Appendix G, ''Fracture Toughness Requirements,'' which requires that the P/T limits for an operating plant be at least as conservative as those that would be generated if the methods of American Society of Mechanical Engineers, Section XI, Appendix G, were applied. The use of an NRC-approved methodology, together with conservatively chosen plant-specific input parameters, provides an acceptable margin of safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the above responses, EGC concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 and, accordingly, a finding of no significant hazards consideration is justified.

The NRC staff has reviewed EGC'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 NSHC.

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

NRC Section Chief: Gene Y. Suh.

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

Date of application for amendments: September 15, 2004.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain containment hydrogen and oxygen monitors. A notice of availability for this technical specification improvement using the consolidated line item improvement process (CLIIP) was published in the **Federal Register** on September 25, 2003 (68 FR 55416). Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.''

Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for combustible gas control system in light-water-cooled power reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the relevant portions of the model NSHC determination (hydrogen and oxygen monitors only) in its application dated September 15, 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 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 and oxygen 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 and oxygen 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. Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen 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, [classification of the oxygen monitors as Category 2,] and removal of the hydrogen and oxygen monitors from TS will not prevent an accident management strategy through the use of the severe accident management guidelines (SAMGs), the emergency plan (EP), the emergency operating procedures (EOPs), 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 and oxygen 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 and oxygen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen and oxygen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen and oxygen 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 and oxygen 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.

Category 2 oxygen monitors are adequate to verify the status of an inerted containment.

Therefore, this change does not involve a significant reduction in the margin of safety. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors. Removal of hydrogen and oxygen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

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

NRC Section Chief: Darrell Roberts.

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

Date of amendment request: September 21, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the relevant portions of the model NSHC determination (hydrogen monitors only) in its application dated September 21, 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 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. Category 1 in RG 1.97 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 severe accident management guidelines (SAMGs), the emergency plan (EP), the emergency operating procedures (EOPs), 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.

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.

NRC Section Chief: Michael L. Marshall.

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

Date of amendment request: January 13, 2005.

Description of amendment request: The proposed change would allow a one-time extended allowed outage time (AOT) change to Improved Technical Specifications (ITS) 3.5.2, Emergency Core Cooling Systems (ECCS)—Operating; 3.6.6, Reactor Building Spray and Containment Cooling Systems; 3.7.8, Decay Heat Closed Cycle Cooling Water System (DC); and 3.7.10, Decay Heat Seawater System to allow the refurbishment of Decay Heat Seawater System Pump RWP–3B online.

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:

This request has been evaluated against the standards in 10 CFR 50.92, and has been determined to not involve a significant hazards consideration. In support of this conclusion, the following analysis is provided:

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

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days. These Systems are designed to provide cooling for components essential to the mitigation of plant transients and accidents. The systems are not initiators of design basis accidents. The proposed ITS changes have been evaluated to assess their impact on normal operation of the systems affected and to ensure that their design basis safety functions are preserved.

A Probabilistic Safety Assessment (PSA) has been performed to assess the risk impact of an increase in Completion Time from 72 hours to 10 days. Although the proposed one-time change results in an increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), the value of these increases are considered as small (CDF) and very small (LERF) in the current regulatory guidance.

Therefore, granting this LAR [License Amendment Request] does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days.

The proposed LAR will not result in changes to the design, physical configuration of the plant or the assumptions made in the safety analysis. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed license amendment extends, on a one-time basis, the Completion Time for the systems described above from 72 hours to 10 days. The proposed change will allow online repair of Decay Heat Seawater pump RWP–3B to restore the pump to full qualification which will improve its reliability and useful lifetime, thus increasing the long term margin of safety of the system.

The proposed LAR will reduce the probability (and associated risk) of a plant shutdown to repair a Decay Heat Services Seawater pump. To ensure defense-in-depth capabilities and the assumptions in the risk assessment are maintained during the proposed one-time extended Completion Time, CR–3 will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities and no maintenance activities of other risk sensitive equipment beyond that required for the refurbishment activity will be scheduled concurrent with the repair activity. Other compensatory actions that will be implemented include: operator attention to the importance of protecting the operable redundant train and support systems will be increased, selection of beneficial Makeup Pump configurations, no elective maintenance will be scheduled in the switchyard, and the establishment of fire watches.

As described above in Item 1, a PSA has been performed to assess the risk impact of an increase in Completion Time. Although the proposed one-time change results in an increase in Core Damage Frequency (CDF), and Large Early Release Frequency (LERF), the value of these increases is considered as small (CDF) and very small (LERF) in the current regulatory guidance.

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

Based on the above, Progress Energy Florida, Inc. (PEF) concludes that the proposed LAR presents a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Michael L. Marshall.

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

Date of amendment request: October 29, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to Operable status within 7 days.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination in its application dated October 29, 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

A change is proposed to allow the affected SDV vent and drain line to be isolated when there are one or more SDV vent or drain lines with one valve inoperable instead of requiring the valve to be restored to operable status within 7 days. With one SDV vent or drain valve inoperable in one or more lines, the isolation function would be maintained since the redundant valve in the affected line would perform its safety function of isolating the SDV. Following the completion of the required action, the isolation function is fulfilled since the associated line is isolated. The ability to vent and drain the SDV is maintained and controlled through administrative controls. This requirement assures the reactor protection system is not adversely affected by the inoperable valves. With the safety functions of the valves being maintained, the probability or consequences of an accident previously evaluated are not significantly increased.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, 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 proposed change ensures that the safety functions of the SDV vent and drain valves are fulfilled. The isolation function is maintained by redundant valves and by the required action to isolate the affected line. The ability to vent and drain the SDV is maintained through administrative controls. In addition, the reactor protection system will prevent filling of the SDV to the point that it has insufficient volume to accept a full scram. Maintaining the safety functions related to isolation of the SDV and insertion of control rods ensures that the proposed change does not involve a significant reduction in the 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: M. Kotzalas (Acting).

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

Date of amendment requests: December 27, 2004.

Description of amendment requests: The requested change will delete Technical Specification (TS) 5.7.1.1.a, ''Occupational Radiation Exposure Report,'' and TS 5.7.1.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 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:

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating letter report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: December 27, 2004.

Description of amendment requests: The proposed amendments would revise the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3 accident source term used in the design basis radiological consequences analyses. These license amendments are requested in accordance with the requirements of 10 CFR 50.67, which addresses the use of an Alternative Source Term (AST) at operating reactors, and relevant guidance of Regulatory Guide 1.183. These license amendments represent full-scope implementation of the AST described in Regulatory Guide 1.183.

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

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

Response: No.

The proposed changes to the Facility Operating Licenses for San Onofre Units 2 and 3 credit an Alternative Source Term (AST) for the design basis radiological site boundary and control room dose analyses. This change represents full scope implementation of the AST as described in Regulatory Guide 1.183. The proposed changes to the Facility Operating Licenses also expand the allowed use of fuel failure estimates by Departure from Nucleate Boiling (DNB) statistical convolution methodology from only the reactor coolant pump sheared shaft event to the Updated Final Safety Analysis Report (UFSAR) Chapter 15 non-Loss-of-Coolant-Accident (LOCA) events that assume a loss of flow (*i.e.*, a loss of AC power) and that fail fuel. The proposed changes reflect the parameters used in the radiological consequences calculations for the LOCA, Fuel Handling Accident inside containment (FHA–IC), Fuel Handling Accident in the Fuel Handling Building (FHA–FHB) and pre-trip Steam Line Break Outside Containment (SLB–OC).

The purpose of this proposed change is to change the design requirements for the Control Room Envelope (CRE). This proposed change will allow an increase in the assumed amount of unfiltered air inleakage through the CRE. Currently, design basis radiological consequence analyses assume CRE inleakage of 0 cfm, plus an assumed 10 cubic feet per minute (cfm) inleakage due to ingress and egress into the Control Room. Analyses to support this change demonstrate acceptable post-accident dose consequences in the Control Room assuming 990 cfm of CRE inleakage (plus 10 cfm due to ingress and egress for a total of 1000 cfm).

This proposed change does not affect the precursors for accidents or transients analyzed in Chapter 15 of the San Onofre Units 2 and 3 UFSAR. Therefore, there is no increase in the probability of accidents previously evaluated. The probability remains the same because the accident analyses performed involve no change to a system, component or structure that affects initiating events for any UFSAR Chapter 15 accident evaluated.

A re-analysis of the UFSAR Chapter 15 LOCA, SLB–OC, FHA–IC, and FHA–FHB events was conducted with respect to radiological consequences. This re-analysis was performed in accordance with AST methodology provided in Regulatory Guide (RG) 1.183 and with ARCON96 atmospheric dispersion methodology provided in RG 1.194. The reanalysis consequences were expressed in terms of Total Effective Dose Equivalent (TEDE) dose.

Implementation of the AST methodology, as described in 10 CFR 50.67, specifies control room, exclusion area boundary (EAB), and low population zone (LPZ) dose acceptance criteria in terms of TEDE dose. The dose acceptance criteria for specific events are specified in RG 1.183. The revised analyses for all evaluated events meet the applicable RG 1.183 TEDE dose acceptance criteria for AST implementation.

The previous dose calculations analyzed the dose consequences to thyroid and whole body as a result of postulated design basis events. The previous control room dose calculations were shown to be within the regulatory limits of 10 CFR 50 Appendix A General Design Criterion 19 with respect to thyroid, beta-skin and whole body dose. The previous LOCA and SLB offsite dose calculations were shown to be within the regulatory limits of 10 CFR 100.11 with respect to thyroid and whole body dose. The previous FHA–IC and FHA–FHB offsite dose calculations were shown to be well within (*i.e.*, less than 25 percent of) the regulatory limits of 10 CFR 100.11 with respect to thyroid and whole body dose. RG 1.183 Footnote 7 provides a means to compare the thyroid and whole body dose results of the previous calculations with the TEDE results of the AST calculations. This methodology requires multiplying the previous thyroid dose by 0.03 and adding the product to the previous whole body dose. The resultant ''effective'' TEDE is then compared to the AST TEDE result. This comparison is presented in Table 5–1.

The Table 5–1 comparison shows a decrease in dose consequences when evaluated using AST methodology for all but the LOCA offsite dose receptors. The LOCA EAB dose using AST methodology has increased due to the requirement to calculate the maximum 2-hour window EAB dose versus the previous requirement to calculate the 0 to 2 hour window EAB dose. The LOCA LPZ dose using AST methodology has increased primarily due to changes in the AST Refueling Water Storage Tank (RWST) iodine transport model. Although the LOCA EAB and LPZ doses using AST methodology have increased, they remain significantly below the 25 Rem TEDE offsite dose acceptance criterion.

TABLE 5–1.—COMPARISON OF PREVIOUS AND AST DOSES

Event-dose receptor	''Effective'' TEDE of previous dose analyses (Rem)	AST TEDE (Rem)
FHA–IC:		
Control Room	1.0	2.7 E–01
EAB	2.0	8.0 E–01
LPZ	5.6 E–02	2.3 E–02
FHA–FHB:		
Control Room	3.7 E–01	7.3 E–02
EAB	6.6 E–01	2.1 E–01
LPZ	1.9 E–02	6.1 E–03
LOCA:		
Control Room	4.5	2.7
EAB	3.7	5.1
LPZ	1.2	1.8
SLB–OC:		
Control Room	(1)	2.1
EAB	8.0	4.1
LPZ	(1)	0.1

[1] Not evaluated.

The proposed changes do not increase the probability of an accident previously evaluated. The proposed changes result in dose consequences that, if compared to previous ones, are in most cases decreased and in other cases only slightly increased (using guidance in footnote 7 of RG 1.183). However, the dose consequences of the revised analyses are below the AST regulatory acceptance criteria.

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

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

Response: No.

The implementation of this proposed change does not create the possibility of an accident of a different type than was previously evaluated in the UFSAR. The proposed change credits the AST for the design basis radiological site boundary and control room dose analyses and expands the allowed use of fuel failure estimates by DNB statistical convolution methodology from only the reactor coolant pump sheared shaft event to the UFSAR Chapter 15 non-LOCA events that assume a loss of flow (*i.e.*, a loss of AC power) and that fail fuel. The changes proposed do not change how Design Basis Accident (DBA) events were postulated nor do the changes themselves initiate a new kind of accident with a unique set of conditions. The changes proposed are based on a re-analysis of offsite and control room doses for four design basis accidents. The revised analyses are consistent with the regulatory guidance established in RG 1.183. The revised analyses utilize the most current understanding of source term timing and chemical forms. Through this re-analysis, no new accident initiator or failure mode was identified.

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

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

Response: No.

The implementation of this proposed amendment does not reduce the margin of safety. The alternative source term radiological dose consequence analyses utilize the regulatory acceptance criteria of 10 CFR 50 Appendix A General Design Criterion (GDC) 19 and 10 CFR 50.67, as specified in RG 1.183. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits. The radiological analyses results remain within these regulatory acceptance criteria.

Therefore, there is no significant reduction in the margin of safety as a result of the proposed amendment.

Based on the above, SCE concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 12, 2004.

Description of amendment request: The proposed amendments would revise Technical Specifications 3.1.7, ''Standby Liquid Control (SLC) System,'' for Hatch Units 1 and 2. The proposed amendments would update Figure 3.1.7–1 of Units 1 and 2 TS to reflect the increased concentration of Boron-10 in the solution. Conforming revisions to Bases B 3.1.7, ''Standby Liquid Control (SLC) System'' are also included.

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?

This is a proposed change to Figure 3.1.7–1 of the Units 1 and 2 Technical Specifications. This figure is a graph of the weight percent of Sodium Pentaborate solution in the Standby Liquid Control (SLC) Tank, as a function of the gross volume of solution in the tank. The figure is proposed to be changed in order to accommodate an injection of Sodium Pentaborate solution into the reactor, following an ATWS event, such that the concentration of Boron-10 atoms in the reactor will be 800 ppm natural Boron equivalent. This is necessary to accommodate increased cycle energy requirements for the Hatch Units 1 and 2 cores.

The proposed change to the Figure will not increase the probability of an ATWS event because the curve has nothing to do with the prevention of an ATWS event. The new requirements will ensure that, in the future, the core will have adequate shutdown margin to mitigate the consequences of an ATWS event.

Also, no systems or components designed to ensure the safe shutdown of the reactor are being physically changed as a result of this proposed TS change. In fact, no safety related systems or components designed for the prevention of previously evaluated events are being altered by the amendment.

As a result, the probability and consequences of an ATWS event, or any other previously evaluated event, will not increase as a result of this amendment.

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

This proposed TS revision results in a change to the SLC TS figure 3.7.1–1 requirements. However, this does not result in physical changes to the SLC system. SLC pump operation, maintenance and testing remain the same. Accordingly, no changes to the operation, maintenance or surveillance procedures will result from this TS revision request. Therefore, no new modes of operation are introduced by this TS change.

Since no new modes of operation are introduced, the proposed change does not create the possibility of a new or different type event from any previously evaluated.

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

This proposed TS change is being made to increase the boron concentration requirements of the sodium pentaborate solution injected into the reactor vessel following an Anticipated Transient Without Scram (ATWS) event. The change is necessary due to new fuel designs and higher energy requirements for fuel cycles. Therefore, the change is being made to insure that shutdown requirements can be met for the ATWS event. This will insure the margin of safety with respect to ATWS will continue to be met.

Consequently, this proposed TS change will not result in a decrease 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: October 14, 2004.

Description of amendment request: The requested change will delete Technical Specification (TS) 6.9.1.5 related to the annual ''Occupational Radiation Exposure Report,'' and TS 6.9.1.10, ''Monthly Reactor Operating Report.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated October 14, 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:

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating letter report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Michael L. Marshall, Jr.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety

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

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina; Carolina Power & Light Company, Docket No. 50–261, H.B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendments: December 19, 2003, as supplemented January 14, 2004.

Brief description of amendments: The amendments allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program as proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359.

Date of issuance: January 11, 2005.

Effective date: January 11, 2005.

Amendment Nos.: 233 and 260.

Facility Operating License Nos. DPR–71, DPR–62, and DPR–23.: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 11, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 26, 2004, as supplemented on December 22, 2004.

Brief description of amendment: The amendment revises Technical Specification 3.7.11, ''Control Room Ventilation System (CRVS),'' to allow, on a one-time basis, an extension of the allowed outage time to support placement of the CRVS in an alternate configuration for tracer gas testing. The proposed amendment would also allow self-contained breathing apparatus and potassium iodide pills to be used as compensatory measures for the control room operators in the event that the tracer gas test results are not bounded by the dose consequence evaluations.

Date of issuance: January 19, 2005.

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

Amendment No.: 223.

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

*Date of initial notice in **Federal Register:*** November 8, 2004 (69 FR 64792).

The December 22 letter provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 25, 2003.

Brief description of amendments: The amendments modify the Limerick Generating Station, (LGS) Units 1 and 2, Technical Specifications (TSs) contained in Appendix A to Operating License Nos. NPF–39 and NPF–85, respectively. The amendments add a footnote to the LGS TS 3.4.3.2.e to indicate that reactor coolant system (RCS) pressure isolation valve leakage is excluded from any other allowable RCS operational leakage specified in LGS TS 3.4.3.2.

Date of issuance: January 18, 2005.

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

Amendment Nos.: 172 and 134.

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

*Date of initial notice in **Federal Register:*** February 3, 2004 (69 FR 5203).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 31, 2004.

Brief description of amendment: This amendment revised Technical Specification (TS) requirements for mode change limitations in Limiting Condition for Operation 3.0.4 and Surveillance Requirement 3.0.4 to adopt the provisions of Industry TS Task Force (TSTF) change TSTF–359, ''Increase Flexibility in Mode Restraints.''

Date of issuance: January 6, 2005.

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

Amendment No.: 131.

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

*Date of initial notice in **Federal Register:*** July 6, 2004 (69 FR 40675).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 19, 2003.

Brief description of amendment: The amendment modifies TS requirements to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, ''Increased Flexibility in Mode Restraints.'' The availability of TSTF–359 for adoption by licensees was announced in the **Federal Register** on April 4, 2003 (68 FR 16579).

Date of issuance: January 11, 2005.

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

Amendment No.: 215.

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

*Date of initial notice in **Federal Register:*** February 17, 2004 (69 FR 7523).

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

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: April 23, 2004.

Brief description of amendments: The amendments revise several Technical Specification (TS) Allowed Outage Times for TS 3.3.3, Accident Monitoring Instrumentation, to be consistent with the Completion Times in the related Specification in NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plants (the Improved Standard Technical Specifications, or ISTS).''

Date of issuance: January 6, 2005.

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

Amendment Nos: 227 and 223.

Renewed Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 25, 2004 (69 FR 29767).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 23, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) requirements to adopt the provisions of the TS Task Force (TSTF) change TSTF–359, regarding increased flexibility in mode changes. The availability of TSTF–359 for adoption by licensees was announced in the **Federal Register** on April 4, 2003 (68 FR 16579).

Date of issuance: January 10, 2005.

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

Amendment No.: 255.

Facility Operating License No. DPR–49: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 16, 2004 (69 FR 55844).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 10, 2005.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: September 30, 2004.

Brief description of amendments: The amendments delete Technical Specification (TS) 6.9.1.2, ''Occupational Radiation Exposure Report,'' and TS 6.9.1.5, ''Monthly Operating Reports,'' as described in the Notice of Availability published in the **Federal Register** on June 23, 2004 (69 FR 35067).

Date of issuance: January 5, 2005.

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

Amendment Nos.: Unit 1–168; Unit 2–157.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 26, 2004 (69 FR 62478).

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

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: February 3, 2004 as supplemented by letter dated December 1, 2004.

Brief description of amendments: The amendments modify Technical Specifications (TSs) requirements to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, ''Increase Flexibility in Mode Restraints.'' The availability of TSTF–359 for adoption by licensees was announced in the **Federal Register** on April 4, 2003 (68 FR 16579).

Date of issuance: January 10, 2005.

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

Amendment Nos.: Unit 1–170; Unit 2–158.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 2, 2004 (69 FR 9865).

The supplement dated December 1, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 10, 2005.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management 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 PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. 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, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov*. If a request for a hearing or petition for leave to intervene is filed by the above date, 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 identify 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 intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner 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.[1] 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 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.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. Miscellaneous—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the

[1] To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

authority to act for the petitioners/ requestors with respect to that contention.

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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 or 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).

STP Nuclear Operating Company, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: January 6, 2005.

Description of amendment request: The amendment revises Technical Specification (TS) 3.7.4, ''Essential Cooling Water System,'' and the associated TS for systems supported by the Essential Cooling Water (ECW), to extend the allowed outage time for an additional 7 days for ECW Train B as a one-time change for the purpose of making repairs to the Train B ECW pump.

Date of issuance: January 10, 2005.

Effective date: Effective as of the date of issuance and shall be implemented immediately.

Amendment No.: 169.

Facility Operating License No. NPF–76: Amendment revises the technical specifications.

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated January 10, 2005.

Attorney for licensee: A.H. Gutterman, Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Michael K. Webb, Acting.

Dated at Rockville, Maryland, this 24th day of January 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 05–1574 Filed 1–31–05; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Notice of Availability of Draft NUREG–1800, Revision 1; ''Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants'' and Draft NUREG–1801, Revision 1; ''Generic Aging Lessons Learned (GALL) Report''

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Issuance of draft NUREG–1800 ''Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants'' and draft NUREG–1801, ''Generic Aging Lessons Learned (GALL) Report'' for public comment; and announcement of public workshop.

SUMMARY: The NRC staff is issuing drafts of the revised NUREG–1800; ''Standard Review Plan for License Renewal Applications for Nuclear Power Plants'' (SRP–LR); and the revised NUREG–1801, ''Generic Aging Lessons Learned (GALL) Report'' for public comment. These revised documents describe methods acceptable to the NRC staff for implementing the license renewal rule, Title 10, Code of Federal Regulations part 54 (10 CFR part 54), as well as techniques used by the NRC staff in evaluating applications for license renewals. The NRC is also announcing a public workshop to facilitate gathering public comments on the drafts of these revised documents. These draft documents supersede the preliminary draft documents that were publicly announced and placed on the NRC's Web site at *http://www.nrc.gov/reactors/operating/licensing/renewal/guidance/updated-guidance.html* on September 30, 2004. The NRC is especially interested in stakeholder comments that will improve the safety, effectiveness, and efficiency of the license renewal process.

DATES: Comments may be submitted on revised SRP-LR and the draft GALL Report, accompanied by supporting data, by March 30, 2005. Comments received after this date will be considered, if it is practical to do so, but the NRC staff is able to ensure consideration only for comments received on or before this date. A public workshop is planned for March 2, 2005, at NRC's headquarters and is announced on the NRC's Web site at *http://www.nrc.gov/public-involve/public-meetings/meeting-schedule.html*.

ADDRESSES: Written comments may be submitted to: Chief Rules and Directives Branch, Division of Administrative Services, Office of Administration, Mailstop T–6D59, U.S. Nuclear Regulatory Commission, Washington, DC, 20555–0001. Comments should be delivered to: 11545 Rockville Pike, Rockville, Maryland, Room T–6D59, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Comments may also be provided via e-mail at *NRCREP@NRC.GOV*. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html*. Persons who do not have access to ADAMS, or who encounter problems in accessing the documents located in ADAMS, should contact the NRC's PDR Reference staff at 1–800–397–4209, or 301–414–4737, or by e-mail to *pdr@nrc.gov*.

FOR FURTHER INFORMATION CONTACT: Mr. Jerry Dozier, License Renewal Project