II. EA Summary

The purpose of the proposed action is to allow for the release of Buildings 122 and 124 and associated outdoor areas at the licensee's New Brunswick, New Jersey facility for unrestricted use. E.R. Squibb & Sons, Inc. was authorized by NRC from 1964 to use radioactive materials for research and development and manufacturing and distribution purposes at the site. On October 16, 2003, E.R. Squibb & Sons, Inc. requested that NRC release Buildings 122 and 124 and associated outdoor areas at the New Brunswick facility for unrestricted use. E.R. Squibb & Sons, Inc. has conducted surveys of the buildings and associated outdoor areas and determined that the buildings and outdoor areas meet the license termination criteria in subpart E of 10 CFR part 20. The NRC staff has prepared an EA in support of the proposed license amendment.

III. Finding of No Significant Impact

The staff has prepared the EA (summarized above) in support of the proposed license amendment to release the buildings for unrestricted use. The NRC staff has evaluated E.R. Squibb & Sons, Inc.'s request and the results of the surveys and has concluded that the completed action complies with the criteria in subpart E of 10 CFR part 20. The staff has found that the environmental impacts from the proposed action are bounded by the impacts evaluated by the ''Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Facilities'' (NUREG–1496). On the basis of the EA, the NRC has concluded that the environmental impacts from the proposed action are expected to be insignificant and has determined not to prepare an environmental impact statement for the proposed action.

IV. Further Information

The EA and the documents related to this proposed action, including the application for the license amendment and supporting documentation, are available for inspection at NRC's Public Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html* (ADAMS Accession No. ML040830086). These documents are also available for inspection and copying for a fee at the Region I Office, 475 Allendale Road, King of Prussia, Pennsylvania 19406. Persons who do not have access to ADAMS should contact the NRC PDR Reference staff by telephone at (800) 397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at King of Prussia, Pennsylvania, this 23rd day of March, 2004.

For the Nuclear Regulatory Commission.

John D. Kinneman,

Chief, Nuclear Materials Safety Branch 2, Division of Nuclear Materials Safety, Region I.

[FR Doc. 04–7011 Filed 3–29–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of March 29, April 5, 12, 19, 26, May 3, 2004.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of March 29, 2004

There are no meetings scheduled for the Week of March 29, 2004.

Week of April 5, 2004—Tentative

There are no meetings scheduled for the Week of April 5, 2004.

Week of April 12, 2004—Tentative

Tuesday, April 13, 2004

9:30 a.m. Briefing on Status of Office of Nuclear Regulatory Research (RES) Programs, Performance, and Plans (Public Meeting) (Contact: Alan Levin, (301) 415–6656). This meeting will be webcast live at the Web address—*www.nrc.gov.*

Week of April 19, 2004—Tentative

There are no meetings scheduled for the Week of April 19, 2004.

Week of April 26, 2004—Tentative

Wednesday, April 28, 2004

9:30 a.m. Discussion of Security Issues (closed—ex. 1).

Week of May 3, 2004—Tentative

Tuesday, May 4, 2004

9:30 a.m. Briefing on Results of the Agency Action Review Meeting (Public Meeting) (Contact: Bob Pascarelli, (301) 415–1245). This meeting will be webcast live at the Web address—*www.nrc.gov.*

Thursday, May 6, 2004

1:30 p.m. Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, (301) 415–7360). This meeting will be webcast live at the Web address—*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.

* * * * *

Additional Information

By a vote of 3–0 on March 16 and 18, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Security Issues (closed—ex. 1 & 2)'' be held March 22, and on less than one week's notice to the public.

By a vote of 3–0 on March 23, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of (1) Private Fuel Storage, LLC (Independent Spent Fuel Storage Installation) Intervenor Ohngo Gaudadeh Devia's Motion to Reopen the Case Record on Contention ''O''—Environmental Justice, and (2) Private Fuel Storage (Independent Spent Fuel Storage Installation) Docket No. 72–22–ISFI'' be held on March 24, and on less than one week's notice to the public.

* * * * *

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

* * * * *

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

Dated: March 25, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–7161 Filed 3–26–04; 9:54 am]

BILLING CODE 7590—01—M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 5, 2004 through March 18, 2004. The last biweekly notice was published on March 16, 2004 (69 FR 12361).

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

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

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

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

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

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

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

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

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

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

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

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

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

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: January 9, 2004.

Description of amendment requests: The Haddam Neck Plant (HNP) is currently undergoing active decommissioning. The proposed amendment would revise Technical Specifications (TS) 6.6.4, 6.7.1, and 6.8 in accordance with Technical Specification Task Force (TSTF) travelers 152, 258 and 308 to reflect changes to Title 10 Part 20 of the Code of Federal Regulations (CFR). The proposed amendment would also revise TS 6.1, 6.2.1, 6.4, 6.5, and 6.6 to reflect the use of generic organizational titles, modeled after TSTF 65 revision 1.

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

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

No. The proposed changes are proposed to reflect the current version of 10 CFR 20 and to eliminate the need for a TS change each time there is a organizational change. These changes do not impact any design basis accidents described in the updated Final Safety Analysis Report (FSAR) for the HNP. Since the proposed changes are administrative or editorial, they cannot affect the likelihood or consequences of accidents.

Therefore, the proposed administrative changes to the Operating License and Technical Specifications will not increase the probability or consequences of an accident previously evaluated.

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

No. The proposed changes do not affect plant system operation. The proposed changes do not involve a physical alteration to the plant or any change in plant configuration. The proposed changes do not require any new operator actions. The changes do not alter the way any structure, system, or component functions. The changes do not introduce any new failure modes.

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

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

No. The proposed changes will make the HNP Operating License and Defueled Technical Specifications consistent with the current 10 CFR 20, and eliminate the need for a TS change each time there is an organizational change. The proposed changes will not result in any technical changes to current requirements. The proposed changes have no effect on assumptions and any acceptance criteria for the design basis accidents described in the updated FSAR for the HNP.

Therefore, the proposed administrative changes will not result in a 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.

NRC Section Chief: Claudia Craig.

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

Date of amendment request: February 9, 2004.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TS) 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 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 model NSHC determination in its application dated February 9, 2004.

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

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

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

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

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations (PARs) 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 PARs.

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: February 9, 2004.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TS) 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 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 (NHSC) 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 February 9, 2004. *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

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

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

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations (PARs) 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 PARs.

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 3, 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 seven 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 February 3, 2004. *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

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 SDVs 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 SDVs is maintained through administrative controls. In addition, the reactor protection system will prevent filling of an 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: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 9, 2003.

Description of amendment request: The amendment would allow a one-time increase in the completion time for restoring an inoperable emergency feedwater (EFW) system train to operable status to allow the realignment of the diesel-driven EFW pump during power operations.

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

1. 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 restoring an inoperable Emergency Feedwater System train to Operable status. The Emergency Feedwater System is designed to provide cooling for components essential to the mitigation of plant transients and accidents. The system is not an initiator of design basis accidents. During the requested extended time period of 14 days, the redundant Emergency Feedwater Pump (EFP) will be available and capable of providing cooling to the Once-Through Steam Generators (OTSGs) during emergency conditions. In addition, a safety-grade motor driven pump (EFP–1) is available for manual initiation and is capable of providing adequate EFW flow for OTSG cooling during all design basis events. EFP–1 is also capable of being supplied by the ''A'' train emergency diesel generator if sufficient electrical loading capacity is available during a loss of offsite power condition. Although Feedwater (FW) pump FWP–7 is non-safety related and its motor is non-seismic, it will also be available and capable of providing OTSG cooling during all but the most limiting design basis events. FWP–7 also has a non-safety diesel backup power supply in the event normal power is not available.

A Probabilistic Safety Assessment (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 and Large Early Release Frequency, the value of these increases are considered as very small 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 restoring an inoperable Emergency Feedwater System train to Operable status.

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 restoring an inoperable Emergency Feedwater System train to Operable status. The proposed change will allow online alignment of one of the Emergency Feedwater pumps to improve its reliability, 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 an Emergency Feedwater 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 [Crystal River Unit 3] will continue the performance of 10 CFR 50.65(a)(4) assessments before performing maintenance or surveillance activities. Other compensatory actions that may be implemented include: use of pre-job briefings and periodic operator walkdowns to assess the status of risk sensitive equipment in the redundant train, use of operator walkdowns to assess and limit transient combustibles in risk significant fire areas, and no elective maintenance to be scheduled in the switchyard that would challenge the availability of offsite power to the ES [engineered safeguards] buses.

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 and Large Early Release Frequency, the value of these increases are considered as very small in the current regulatory guidance.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, 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: William F. Burton, Acting.

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 amendment request: March 3, 2004.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Surveillance Requirement 4.0.5 by updating the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code references as the source of inservice testing requirements for ASME Code Class 1, 2, and 3 pumps and valves. The proposed amendments replace reference to Section XI of the Code with reference to ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

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

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

The proposed changes do not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed changes. The amendments application proposes to revise the Turkey Point Units 3 and 4 Technical Specifications Surveillance Requirement 4.0.5. The proposed changes would revise the technical specifications to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves for the Fourth 10-Year interval.

The current Turkey Point Units 3 and 4 Technical Specifications reference the ASME Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code, which is consistent with 10 CFR Section 50.55a(f). In addition, surveillance interval definitions for ''biennially or every 2 years'' as used in the ASME OM Code would be added to TS surveillance requirement 4.0.5.b to ensure consistent interpretation of the terms.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no new or different accident initiators are introduced by these proposed changes.

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

The proposed changes do not involve a significant reduction in a margin of safety because there are no changes to initial conditions contributing to accident severity or consequences. Thus, there is not 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: December 23, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to eliminate the reactor head cooling containment isolation function since the reactor head cooling system has been removed from service. In addition, the TS are being changed to correct and clarify existing requirements, make wording enhancements, and revise an existing limiting condition for operation for radiation monitors used to isolate reactor building ventilation and initiate the standby gas treatment system (SGTS).

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

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

Response: No.

One of the proposed changes removes the Reactor Head Cooling system primary containment isolation signal from the TS. The existing piping will be removed and the existing process pipe through the containment penetration will be cut and capped. This equipment was only used for the shutdown-cooling (non-safety related) mode of operation. This system does not support safe shutdown of the facility. The proposed TS change does not introduce new equipment or new equipment operating modes, nor does the proposed change alter existing system relationships. These proposed changes do not increase the likelihood of the malfunction of any structure, system or component (SSC) or impact any analyzed accident. Consequently, the probability of an accident previously evaluated is not increased.

The other proposed change adds an allowable outage time to the radiation monitors described in TS that initiate the SGTS and adds a time requirement for placing inoperable channels in a tripped condition. The proposed TS change does not introduce new equipment or new equipment operating modes, nor does the proposed change alter existing system relationships. The change does not affect plant operation, design function or any analysis that verifies the capability of a SSC to perform a design function. Further, the proposed change does not increase the likelihood of the malfunction of any structure, system or component (SSC) or impact any analyzed accident. Consequently, the probability of an accident previously evaluated is not affected.

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

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

Response: No.

One of the proposed changes removes the Reactor Head Cooling system primary containment isolation signal from the TS. The existing piping will be removed and the existing process pipe through the containment penetration will be cut and capped. This equipment was only used for the shutdown-cooling (non-safety related) mode of operation. The change does not create the possibility of new credible failure mechanisms, or malfunctions. The proposed change does not introduce new accident initiators. Consequently, the changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

The other proposed change adds an allowable outage time to the radiation monitors described in TS that initiate the SGTS and adds a time requirement for placing inoperable channels in a tripped condition. This change does not modify the design function or operation of any SSC. Further the change does not involve physical alterations of the plant; no new or different type of equipment will be installed. The proposed change is not an indicator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not affected.

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

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

One of the proposed changes removes the Reactor Head Cooling system primary containment isolation signal from the TS. The existing piping will be removed and the existing process pipe through the containment penetration will be cut and capped. This equipment was only used for the shutdown-cooling (non-safety related) mode of operation. This system does not support safe shutdown of the facility. This change does not exceed or alter a design basis or a safety limit for a parameter established in the MNGP [Monticello Nuclear Generating Plant] Updated Safety Analysis Report (USAR) or the MNGP facility license. Consequently, the change does not result in a significant reduction in the margin of safety.

The other proposed change adds an allowable outage time to the radiation monitors described in TS that initiate the SGTS and adds a time requirement for placing inoperable channels in a tripped condition. This change ensures continued compliance with regulatory and licensing requirements. The change does not exceed or alter a design basis or safety limit for a parameter established in the MNGP USAR or MNGP facility license. Consequently, the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: February 13, 2004.

Description of amendment requests: The amendment would revise Technical Specifications (TSs) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' and 3.3.6, ''Containment Ventilation Isolation Instrumentation.'' The purpose of the amendment is to adopt the completion time, test bypass time, and surveillance frequency time changes approved by the NRC in Topical Reports WCAP–14333–P–A, ''Probabilistic Risk Analysis of the RPS [reactor protection system] and ESFAS Test Times and Completion Times,'' and WCAP–15376–P–A, ''Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times.'' The proposed changes would revise the required actions for certain action conditions; increase the completion times for several required actions (including some notes); delete notes in certain required actions; increase frequency time intervals (including certain notes) in several surveillance requirements (SRs); add an action condition and required actions; add or revise notes in certain SRs; and revise Table 3.3.1–1. There are also administrative corrections to the format of the TSs (*e.g.*, remove the bold appearance of page number 3.3–45).

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same RTS and ESFAS instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the TS [in the amendment] do not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered.

The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators [because the proposed changes are not event initiators]. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the Updated Final Safety Analysis Report [for Diablo Canyon Units 1 and 2].

The determination that the results of the proposed changes are acceptable [to be considered for plant-specific TS] was established in the NRC Safety Evaluations prepared for WCAP–14333–P–A, Revision 1, (issued by letter dated July 15, 1998) and for WCAP–15376–P–A, Revision 1, (issued by letter dated December 20, 2002). Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions [for the two WCAPs].

The proposed changes to the CTs [completion times], test bypass times, and Surveillance Frequencies reduce the potential for inadvertent reactor trips and spurious engineered safety features actuations, and therefore do not increase the probability of any accident previously evaluated. The proposed changes do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the increase in core damage frequency (CDF) is less than 1.0E–06 per year and the increase in large early release frequency (LERF) is less than 1.0E–07 per year. In addition, for the CT changes, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–07 and 5.0E–08, respectively. These changes meet the acceptance criteria in Regulatory Guides (RGs) 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their [safety] functions with high reliability as originally assumed, and the increase in risk as measured by ΔCDF, ΔLERF, ICCDP, ICLERP risk metrics is within the acceptance criteria of existing [NRC] regulatory guidance, there will not be a significant increase in the consequences of any accidents.

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

Therefore, [the] change[s do] not increase the probability or consequences of any accident previously evaluated.

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

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

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

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

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

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit. There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling limits, local power peaking factor (F_Q), hot channel factor (FΔH), loss-of-coolant accident (LOCA) peak cladding temperature, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan will continue to be met.

Redundant RTS and ESFAS trains are maintained, and diversity with regard [to] the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in RGs 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased CTs and bypass test times, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

(a) Reduced testing will result in fewer inadvertent reactor trips, less frequent actuation of ESFAS components, less frequent distraction of operations personnel without significantly affecting RTS and ESFAS reliability.

(b) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to instrumentation Required Actions with short CTs.

(c) Longer repair times associated with increased CTs will lead to higher quality repairs and improved reliability.

(d) The CT extensions for the reactor trip breakers will provide additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker CT, and provide consistency with the CT for the logic trains.

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

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

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

NRC Section Chief: Stephen Dembek.

Southern California Edison Company, et al., Docket No. 50–206, San Onofre Nuclear Generating Station, Unit 1, San Diego County, California

Date of amendment request: January 28, 2004.

Description of amendment request: Southern California Edison (SCE) permanently shutdown San Onofre Nuclear Generating Station (SONGS), Unit 1, in November 1992. Active decommissioning of SONGS Unit 1 began in June 1999. As part of decommissioning, SCE constructed an Independent Spent Fuel Storage Installation (ISFSI) at SONGS for dry cask storage of spent fuel. In March 2004, SCE plans to begin moving the spent fuel located in the Unit 1 spent fuel pool into the ISFSI. SCE has proposed to eliminate License information and technical specifications which will no longer be applicable following the transfer of the last fuel assembly from the Unit 1 spent fuel pool to the ISFSI.

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?

No. This proposed change provides the necessary requirements for Unit 1 with no spent fuel located in the spent fuel pool. With no spent fuel located at Unit 1, the probability and consequence of the fuel handling accident are removed.

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

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

No. These changes provide the necessary requirements for SONGS Unit 1 with no spent fuel in the spent fuel pool. With no spent fuel located at Unit 1, there is no possibility of a new or different kind of accident.

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

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

No. These changes provide the necessary requirements for SONGS Unit 1 with no spent fuel in the spent fuel pool. With no spent fuel located at Unit 1, the fuel handling accident is not applicable and there is impact on the margin of safety.

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

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

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

NRC Section Chief: Mark Thaggard.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 395, Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1, Fairfield County, South Carolina

Date of amendment request: September 19, 2003.

Description of amendment request: The proposed change will revise the Technical Specifications (TSs) Surveillance Requirement (SR) 4.2.4.2, to reflect the use of the Power Distribution Monitoring System (PDMS) for a core power distribution measurement. This change will also result in revising the Bases for 3/4.2.4 to reflect the use of the PDMS.

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

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

No. The proposed change to TS 4.2.4.2 clarifies the use of the PDMS as means of measuring core power distribution with one Power Range Channel inoperable to determine if QPTR [Quadrant Power Tilt Ratio] is within the limit. The use of PDMS was approved in Amendment 142 and added as TS 3.3.3.11. This clarification of its use in TS 4.2.4.2 specifies an additional method of performing the surveillance requirement and will not increase the probability of an accident previously evaluated.

The probability or consequences of accidents previously evaluated in the VCSNS FSAR [Final Safety Analysis Report] are unaffected by this proposed change because there is no change to any equipment response or accident mitigation scenario. There are no additional challenges to fission product barrier integrity. 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 different kind of accident from any previously evaluated?

No. The proposed change to TS 4.2.4.2 clarifies the use of the PDMS as means of measuring core power distribution with one Power Range Channel inoperable to determine if QPTR is within the limit. The use of the PDMS was approved in Amendment 142 and added as TS 3.3.3.11. This clarification of its use in TS 4.2.4.2 specifies an additional method of performing the surveillance requirement and does not create the possibility of a new different kind of accident or malfunction.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. A change to the surveillance requirement is proposed; however, this clarification of the use of PDMS in TS 4.2.4.2 specifies an additional method of performing the surveillance requirement.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 10, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specification allowable value for the spent fuel pool area radiation monitors.

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

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

No. The proposed technical specification (TS) change to reduce the allowable value for the spent fuel pool area radiation monitors does not change any operator actions nor does it change plant systems or structures. Therefore, the proposed change does not result in a significant increase in the probability of a Fuel Handling Accident (FHA). The surveillance requirement radiation limit for the spent fuel pool area radiation monitors will be lowered to compensate for the change in source terms which resulted from the methodology change due to discovery of a modeling error. This change ensures the monitors perform their safety function of limiting the site boundary dose to a small fraction of the 10 CFR part 100 limits. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed TS change does not alter the function of the spent fuel monitors which is to initiate ABGTS [Auxiliary Building Gas Treatment System actuation] upon an FHA. The TS allowable value and the associated setpoints for the spent fuel pool area radiation monitors will be lowered due to calculation methodology changes resulting from discovery of a modeling error. The change will not result in the installation of any new equipment or system. No new operations procedures, conditions, or modes will be created by this proposed 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 margin of safety?

No. The method for calculating the radiological consequences are revised for calculating the safety limit of the spent fuel pool area radiation monitors to correctly account for isotopic release fractions. The monitors' setpoints are based on 30 rem thyroid at the site boundary resulting from an unfiltered release. At the monitor setpoint, the monitors initiate ABGTS and thus the release is filtered. The radiological dose consequences do not change and remain less than a small fraction of the dose limit identified in 10 CFR 100. The surveillance requirement is being reduced for consistency with calculation methodology changes and to ensure the monitors perform their intended design function of limiting the site boundary dose to less than 30 rem thyroid subsequent to an FHA. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: William F. Burton, Acting.

Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit No. 2, Louisa County, Virginia

Date of amendment request: January 23, 2004.

Description of amendment request: The proposed amendment would revise Improved Technical Specifications (TS) Surveillance Requirements 3.5.1.4, 3.5.4.3, and 3.6.7.3 to delete a note that differentiates between the amount of boron concentrations at North Anna Power Station, Units 1 and 2.

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

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

The proposed changes to TS Surveillance Requirements 3.5.1.4, 3.5.4.3, and 3.6.7.3 delete a note that is no longer necessary and do not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. The proposed changes will not alter assumptions relative to the mitigation of an accident or transient event.

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

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

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

The proposed changes do not alter the boron concentrations in the safety injection accumulators, RWST [refueling water storage tank], and casing cooling tank. The proposed changes to TS Surveillance Requirements 3.5.1.4, 3.5.4.3, and 3.6.7.3 are considered administrative in nature. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of application for amendment: December 3, 2003, as supplemented January 14 and February 6, 2004.

Brief description of amendment: The amendment eliminates a license condition that limits HBRSEP2 operation to 504 effective full-power days. This license condition was added in License Amendment No. 196, issued on November 5, 2002.

Date of issuance: March 10, 2004.

Effective date: March 10, 2004.

Amendment No. 200.

Facility Operating License No. DPR–23: Amendment revises Appendix B, ''Additional Conditions,'' to the Facility Operating License.

Date of initial notice in **Federal Register:** February 3, 2004 (69 FR 5201). The February 6, 2004, supplemental letter provided clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 22, 2003.

Brief description of amendment: The amendment deletes requirements from the Technical Specifications to maintain hydrogen recombiners and hydrogen and oxygen monitors.

Date of issuance: March 15, 2004.

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

Amendment No.: 159.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 3, 2004 (69 FR 5202).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 7, 2003 as supplemented September 18, 2003.

Brief description of amendment: The amendment changed Technical Specifications (TSs) affecting cycle-specific parameters that will be relocated to the Core Operating Limits Report.

Date of issuance: March 9, 2004.

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

Amendment No.: 218.

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

Date of initial notice in ***Federal Register:*** May 27, 2003 (68 FR 28849). The September 18, 2003 supplement contained clarifying information and did not change the staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 14, 2002, supplemented by letters dated September 11, 2003, and March 10, 2004.

Brief description of amendments: The amendments revised the Technical Specification 3.3.2, ''Engineered Safety Features Actuation System Instrumentation.''

Date of issuance: March 16, 2004.

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

Amendment Nos.: 220 & 202.

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

Date of initial notice in ***Federal Register:*** August 19, 2003 (68 FR 49815).

The supplements dated September 11, 2003, and March 10, 2004, provided clarifying information that did not change the scope of the November 14, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 16, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 19, 2003.

Brief description of amendment: The amendment deletes the requirements from the Technical Specifications to maintain hydrogen recombiners and hydrogen analyzers.

Date of issuance: March 9, 2004.

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

Amendment No.: 192.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 20, 2004 (69 FR 2741).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 18, 2003.

Brief description of amendments: The amendments revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the change modifies TS Table 3.3.6.1–1, ''Primary

Containment Isolation Instrumentation,'' to add the requirement to perform a Channel Check in accordance with Surveillance Requirement (SR) 3.3.6.1.1 to thirteen listed instrument functions. The change is the result of the replacement of existing plant equipment with equipment that has the capability of permitting the performance of a Channel Check with the plant in MODES 1, 2, and 3. The change is consistent with the wording specified in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001.

Date of issuance: March 5, 2004.

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

Amendment Nos.: 166/152.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 10, 2003 (68 FR 34667).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: December 22, 2003, as supplemented by letter dated February 13, 2004. The February 13, 2004, submittal provided clarifying information and did not change the staff's proposed finding of no significant hazards.

Brief description of amendment: This amendment revised the safety limit minimum critical power ratio value in TS 2.1 with the reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated core flow from the current specification of 1.10 to 1.07 for two recirculation-loop operation and from 1.11 to 1.08 for single recirculation-loop operation.

Date of issuance: March 12, 2004.

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

Amendment No. 170.

Facility Operating License No. NPF–39. This amendment revised the Technical Specifications.

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

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: February 27, 2003, as supplemented by letters dated April 11 and August 5, 2003.

Brief description of amendments: The amendments revise the Technical Specifications to allow a one-time change in the containment Type A integrated leakage rate test interval that extends the test interval from 10 to 15 years.

Date of issuance: March 8, 2004.

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

Amendment Nos.: 220/214.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 1, 2003 (68 FR 15759). The April 11 and August 5, 2003, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois

Date of application for amendment: November 14, 2003, as supplemented by letters dated December 23, 2003, and January 7, 2004.

Brief description of amendment: The amendment revises the values and wording of the Technical Specifications safety limit minimum critical power ratio (SLMCPR).

Date of issuance: March 10, 2004.

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

Amendment No.: 215.

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

*Date of initial notice in **Federal Register:*** January 20, 2004 (69 FR 2743). The December 23, 2003, and January 7, 2004, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 21, 2003.

Brief description of amendments: These amendments allow transfer of the requirements of Technical Specifications (TSs) 6.5 (Review and Audit), 6.8.2 and 6.8.3 (Procedures and Programs Review Specifics), and 6.10 (Record Retention) to the St. Lucie Plant's Quality Assurance Plan (a licensee-controlled document).

Date of Issuance: March 11, 2004.

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

Amendment Nos.: 189 & 133.

Renewed Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the TSs.

*Date of initial notice in **Federal Register:*** January 6, 2004 (69 FR 698).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 2004.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendments: July 1, 2003, as supplemented by letters dated November 17 and December 22, 2003.

Brief description of amendments: The amendment revised the values of the Safety Limit for Minimum Critical Power Ratio in the Unit 1 Technical Specifications (TSs) 2.1.1.2, clarified fuel design features in TS 4.2.1, and updated the references used to determine the core operating limits in TS 5.6.5.b.

Date of issuance: March 9, 2004.

Effective date: As of the date of issuance and shall be implemented upon startup following the thirteenth refueling and inspection outage.

Amendment Nos.: 216.

Facility Operating License No. NPF–14: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 5, 2003 (68 FR 46245).

The supplements dated November 17 and December 22, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 3, 2003, as supplemented February 9, 2004.

Brief description of amendments: The amendments revised the Technical Specifications. to add a Limiting Condition for Operation (LCO) for the Liner Heat Generation Rate. The new LCO is included in Section 3.2, Power Distribution Limits. The proposed amendments would also change the recirculation loop LCO, Section 5.6.5, and the appropriate Bases.

Date of issuance: March 8, 2004.

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

Amendment Nos.: 239 / 182.

Renewed Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 12, 2003 (68 FR 64128).

The supplement dated February 9, 2004, provided clarifying information that did not change the scope of the October 3, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 18, 2003.

Brief description of amendments: The amendments modified Technical Specification 3.9.6 to correct completion times of ACTIONS B.2 and B.3, which were overlooked in Amendment No. 105.

Date of issuance: March 5, 2004.

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

Amendment Nos.: 110 and 110.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 3, 2004 (69 FR 5209).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 18, 2003, as supplemented by letter dated August 14, 2003.

Brief description of amendments: The amendments modified Technical Specifications (TS) to permanently except seven containment isolation valves in each unit, in residual heat removal and containment spray systems, from local leakage rate testing requirements of 10 CFR Part 50, Appendix J.

Date of issuance: March 5, 2004.

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

Amendment Nos.: 111 and 111.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 15, 2003 (68 FR 8289).

The August 14, 2003, supplemental letter provided clarifying information and did not change the scope of the original **Federal Register** notice or staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 27, 2003, as supplemented by letter dated December 12, 2003.

Brief description of amendment: The amendment (1) revises the definition of dose equivalent radioiodine 131 (I–131), and (2) increases the maximum allowed closure time of each main feedwater isolation valve (MFIV) from 5 seconds to 15 seconds. A plant modification would replace the electro-hydraulic MFIV actuators with system-medium actuators to improve MFIV reliability and reduce maintenance requirements, and the MFIV stroke time would be increased. A plant modification would also replace swing check valves in each auxiliary feedwater (AFW) motor-driven pump discharge line with an automatic recirculation control check valve to reduce the potential for vibration and increase AFW flow margin. The NRC also approves the re-analysis of the steam generator tube rupture with overfill accident submitted in the application.

Date of issuance: March 11, 2004.

Effective date: March 11, 2004, and shall be implemented prior to the entry into Mode 3 in the restart of the Callaway Plant from the Refueling Outage (RO) 13, which is scheduled for April 2004.

Amendment No.: 159.

Facility Operating License No. NPF–30: The amendment revises the Technical Specifications and updates the Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43394).

The additional information provided in the supplemental letter does not expand the scope of the application as noticed and does not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 19th day of March, 2004.

For the Nuclear Regulatory Commission.

Edwin M. Hackett,

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

[FR Doc. 04–6682 Filed 3–29–04; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

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

Regulatory Guide 1.200, ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed