materials for medical research, diagnosis, and therapy purposes and on August 17, 1962 to operate a research reactor in Building 40 at the site. On March 9, 2004, WRAMC requested that NRC release the facility for unrestricted use. WRAMC has conducted surveys of the facility and determined that the facility meets 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 Building 40 in its entirety of the WRAMC facility at 6900 Georgia Avenue, NW., Washington, DC for unrestricted use. The NRC staff has evaluated WRAMC's request and the results of the surveys, performed independent measurements to confirm the results, 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. ML041380084). 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 1–800–397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov*.

Dated at King of Prussia, Pennsylvania this 18th day of May, 2004.

For the Nuclear Regulatory Commission.

Ronald R. Bellamy,

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

[FR Doc. 04–11756 Filed 5–24–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of May 24, 31, June 7, 14, 21, 28, 2004.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of May 24, 2004

Tuesday, May 25, 2004

2 p.m. Discussion of Management Issues (Closed—Ex. 2)

Wednesday, May 26, 2004

10:30 a.m. All Employees Meeting (Public Meeting)

All Employees Meeting (Public Meeting)

Week of May 31, 2004—Tentative

Wednesday, June 2, 2004

9:30 a.m. Briefing on Equal Employment Opportunity Program (Public Meeting) (Contact: Corenthis Kelley, 301–415–7380)

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

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

This meting will be webcast live at the Web address—*www.nrc.gov*

Week of June 7, 2004—Tentative

Thursday, June 10, 2004

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

Week of June 14, 2004—Tentative

There are no meetings scheduled for the Week of June 14, 2004.

Week of June 21, 2004—Tentative

There are no meetings scheduled for the Week of June 21, 2004.

Week of June 28, 2004—Tentative

There are no meetings scheduled for the Week of June 28, 2004.

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 215–1292. Contact person for more information: Dave Gamberoni, (301) 415–1651.

* * * * *

Additional Information

By a vote of 3–0 on May 14 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)'' be held May 20, and on less than one week's notice to the public.

By a vote of 3–0 on May 19 and 20, the Commission determined pursuant to U.S.C 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of (1) Nuclear Fuel Services, Inc. (Erwin, Tennessee); Appeal of LBP–04–05, the Presiding Officer's Ruling on Hearing Requests; (2) Hydro Resources, Inc. (Rio Rancho, New Mexico) Petitions for Review of LBP–04–03 (Financial Assurance); (3) Louisiana Energy Services, L.P. (National Enrichment Center); and (4) Final Rule to amend 10 CFR Part 2, Subpart J, in Regard to the Licensing Support Network'' be held on May 20, 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: May 20, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–11852 Filed 5–21–04; 9:35 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, April 30, through May 13, 2004. The last biweekly notice was published on May 11, 2004 (69 FR 26184).

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.*

Duke Energy Corporation, et al., Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: March 23, 2004.

Description of amendment request: The amendments would revise Technical Specification 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to extend the allowable inspection interval to 20 years.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 24, 2003 (68 FR 37590), on possible amendments to extend the inspection interval for reactor coolant pump (RCP) flywheels, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 22, 2003 (68 FR 60422). The licensee affirmed the applicability of the model NSHC determination in its application dated March 23, 2004.

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

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

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

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

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

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

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed change will not result in plant operation in a configuration outside of the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in RG 1.174. There are no significant mechanisms for inservice degradation of the RCP flywheel. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Stephanie M. Coffin, Acting.

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

Date of amendment request: April 19, 2004.

Description of amendment request: The proposed change revises Limiting Condition for Operation (LCO) 3.7.3, ''Control Room Emergency Filtration System,'' to provide specific conditions and required actions that address degraded control room boundary.

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

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

The proposed Technical Specifications (TS) change involves the Control Room Emergency Filtration (CREF) System and associated control room boundary, which provide a radiological controlled environment from which the plant can be operated following a design basis accident (DBA). The CREF system and the control room boundary are not assumed to be initiators of any analyzed accident and do not affect the probability of accidents. The proposed change adds a Note to LCO 3.7.3 that allows the control room boundary to be opened intermittently under administrative controls. A new Condition B is also added to LCO 3.7.3 to specify a Completion Time of 24 hours to restore an inoperable control room boundary to OPERABLE status before requiring the plant to perform an orderly shutdown. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and Energy Northwest's commitment to implement, via administrative controls, appropriate compensatory measures consistent with the intent of 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. These compensatory measures will serve to minimize the consequences of an open control room boundary and ensure the CREF system can continue to perform its function. As such, these changes will not affect the function or operation of any other systems, structures or components. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change adds a Note to LCO 3.7.3 that allows the control room boundary to be opened intermittently under administrative controls. A new Condition B is also added to LCO 3.7.3 to specify a Completion Time of 24 hours to restore an inoperable control room boundary to OPERABLE status before requiring the plant to perform an orderly shutdown. The CREF system and the control room boundary are designed to protect the habitability of the control room. The CREF system and the control room boundary are not accident initiators and do not affect the probability of accidents. This change is administrative in nature and does not involve any physical changes to the plant. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change adds a Note to LCO 3.7.3 that allows the control room boundary to be opened intermittently under administrative controls. A new Condition B is also added to LCO 3.7.3 to specify a Completion Time of 24 hours to restore an inoperable control room boundary to OPERABLE status before requiring the plant to perform an orderly shutdown. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and Energy Northwest's commitment to implement, via administrative controls, appropriate compensatory measures consistent with the intent of 10 CFR 50, Appendix A, GDC 19. These compensatory measures will serve to minimize the consequences of an open control room boundary and assure that the CREF system can continue to perform its function. Therefore, the proposed TS change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 21, 2003, as supplemented February 10, 2004.

Description of amendment request: The amendment would modify the Technical Specifications (TSs) to delete TS 3.6.4.4, ''Shield Building Annulus Mixing System,'' in its entirety, revise the Main Steam Isolation Valve (MSIV) leakage limits contained within TS Surveillance Requirement 3.6.1.3.10, and delete reference to TS 3.6.4.4 within TS 3.10.1, ''Inservice Leak and Hydrostatic Testing Operation.''

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

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

Response: No.

As discussed above, the proposed changes are to delete the annulus mixing function and deletion of the single MSIV leakage rate limit. A review of the safety analysis report indicates that operation (or mis-operation) of the annulus mixing system, or any component of the annulus mixing system is not considered an initiator of any accident evaluated in the Updated Safety Analysis Report. The deletion of the single MSIV leakage limit of 50 scfh in effect establishes a maximum leakage limit of 150 scfh which is the current total MSIV leakage limit. The elimination of the single MSIV acceptable leakage rate limit does not impact any event initiator. As the proposed changes do not involve any accident initiators, there is no increase in the probability of an accident previously evaluated.

The annulus mixing system and the main steam isolation valves operate following an LOCA [loss-of-coolant accident] to mitigate the consequences of an accident. Elimination of the annulus mixing system and the single MSIV leakage limit will lead to some increase in the dose consequences of a LOCA. The current LOCA dose consequences evaluation for RBS was revised to account for the elimination of the annulus mixing system and for increasing the single MSIV leakage to 150 scfh (applying the total MS–PLCS Division limit to the single MSIV). The results of the revised evaluation with the proposed changes show an increase in the calculated dose consequences, however, the calculated doses were still within the acceptance limits of 10 CFR 50.67. Thus, while there is an increase in the dose consequences of an accident previously identified, the increase is not deemed to be significant.

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

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

Response: No.

The proposed change does not add any equipment, nor is any equipment replaced with equipment with different performance characteristics. Thus, no new initiators are added, and therefore, no new accident types are created as a result of this change. The proposed changes affect performance characteristics assumed in the LOCA dose consequences evaluation, however, the nature of the accidents evaluated in the safety analysis report are not changed.

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

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

Response: No.

With respect to dose consequences for the LOCA event, the margin of safety is considered to be that provided by meeting the 10 CFR 50.67 limits. The revised dose consequences evaluation, which includes the proposed changes, continues to demonstrate that the doses at the exclusion area boundary, the low population zone, and the control room are within the acceptance limits in 10 CFR 50.67. Therefore, there is no reduction in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 16, 2004.

Description of amendment request: The amendment would change Technical Specification (TS) 3.6.5.1.3, regarding drywell bypass leakage testing (DWBT). The change would allow for a one-time extension of the interval (from 10 to 15 years) for performance of the next DWBT.

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

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

Response: No.

The proposed amendment to TS SR 3.6.5.1.3 adds a one-time extension to the current interval for the DWBT. The current interval of ten years, based on past performance, would be extended on a one-time basis to 15-years from the date of the last test. The proposed extension to the DWBT cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since analysis has shown that, the proposed extension of the DWBT frequency has a minimal impact on plant risk. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed extension to the interval for the DWBT does not involve any design or operational changes that could lead to a new or different kind of accident from any accidents previously evaluated. The tests are not being modified, but are only being performed after a longer interval. 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. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

An evaluation of extending the DWBT surveillance frequency from once in 10 years to once in 15 years has been performed using methodologies based on the ILRT [integrated leak rate testing] methodologies. This evaluation assumed that the DWBT frequency was being adjusted in conjunction with the ILRT frequency. This analysis used realistic, but still conservative, assumptions with regard to developing the frequency of leakage classes associated with the DWBT. The results from this conservative analysis indicates that the proposed extension of the DWBT frequency has a minimal impact on plant risk and 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: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 7, 2004.

Description of amendment request: The proposed changes will revise the Waterford 3 Technical Specifications (TS) to clarify the actions of TS 3.4.5.1, Reactor Coolant System (RCS) Leakage; some of the surveillance requirements (SRs) of TS 3.4.5.2, RCS Operational Leakage; and delete duplication in TS 3.3.3.1, Radiation Monitoring Instrumentation. The proposed change is based on NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants,'' Revision 2, dated April 30, 2001. Also, the proposed change will delete the containment atmosphere gaseous radioactivity monitoring system from the TS because this monitor does not meet the requirements of Regulatory Guide 1.45, Revision 0, ''Reactor Coolant Pressure Boundary Leakage Detection Systems,'' and Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Appendix A, General Design Criteria 30, ''Quality of Reactor Coolant System Pressure Boundary.''

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

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

Response: No.

The proposed revisions do not involve any physical change to plant design. The less restrictive changes proposed in this amendment request include relocation of information to the UFSAR [updated final safety analysis report], addition of a TS 3.0.4 exception, utilization of the diversity and redundancy of the Waterford 3 leakage detection instrumentation, allowing diversity in the contingency actions, deletion of SRs, and addition of an allowed outage time when two of three required leakage detection instrumentation is inoperable. The less restrictive changes will not affect the capability of Waterford 3 to detect RCS leakage. At least one RCS leakage detection instrumentation is always required to remain operable, and other leakage detection indication, while not credited specifically for RCS leakage detection, is still available and required to be operable per other TS requirements (*i.e.*, Containment Temperature and Containment Pressure). Also contingency actions are required (*i.e.*, RCS Inventory Balance, containment grab samples, flow switch verification) when any of the RCS leakage detection instrumentation is inoperable. Performance of the RCS inventory balance is the most accurate method of determining and quantifying leakage. The RCS inventory balance is being added as a contingency and replacement for monitoring instrumentation that has continuous indication and alarms in the control room.

The more restrictive changes proposed by this revision do not adversely affect the capability of Waterford 3 RCS leakage detection instrumentation to detect RCS leakage. The deletion of the containment atmosphere gaseous radioactivity monitor is considered a more restrictive change. This monitor does not meet the leakage detection requirements of Regulatory Guide 1.45 and does not meet the requirements for retention specified in 10 CFR 50.36. Deletion of this monitor will reduce the diversity of the Waterford 3 instrumentation for monitoring the containment atmosphere and require the plant to enter an Action statement when the containment atmosphere particulate monitor is inoperable. Requiring performance of an RCS inventory balance when the containment sump monitor is inoperable provides contingency actions when the plant is in a degraded RCS leakage detection condition.

The administrative changes proposed by this revision do not adversely affect the capability of Waterford 3 RCS leakage detection instrumentation to detect RCS leakage. Relocating the requirements associated with the RCS Leak Detection System from various TS to Specification 3.4.5.1 and adding requirement to shutdown when all required RCS leakage detection instrumentation are inoperable are administrative in nature. The relocation of information from one TS to another consolidates information and causes less contusion in the control room by having all requirements for the leakage detection instrumentation in one TS. The addition of a specific action to shutdown when all three leakage detection instrumentation are inoperable versus an implied requirement to enter TS 3.0.3 is being performed to be similar to the STS [Standard Technical Specifications].

None of the above less restrictive, more restrictive, or administrative changes affects the accident analyses. Since the proposed changes only affect the requirements for the detection of RCS leakage, the probability that an accident previously evaluated will occur remains unchanged. The proposed changes do not prevent nor limit the diversity of acceptable detection of RCS leakage. These changes also do not affect the mitigation capability of any accident previously evaluated. The consequences of an accident previously evaluated are not affected since the mitigation of previously evaluated accidents is not affected and leak rate information will remain available to station personnel.

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

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

Response: No.

The aforementioned revisions do not involve any physical change to plant design. None of the proposed changes affect[s] the accident analyses. The RCS water inventory balance is more accurate than normal leak detection methods in regard to actual RCS leak rates, and therefore is an excellent alternative when other leak detection components may become inoperable. The proposed changes do not prevent acceptable detection of RCS leakage by diverse methods. The detection of a RCS leak can not cause an accident. Likewise, detecting a RCS leak, while in its beginning stages, does not create the possibility of a new or different kind of accident than any previously analyzed. Therefore, a new or different kind of accident than that previously analyzed does not result due to the proposed changes of this submittal.

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

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

Response: No.

The aforementioned revisions do not involve any physical change to plant design. The proposed changes do not adversely affect the ability of the RCS leakage detection system to detect RCS leakage. The ability of the RCS leakage detection instrumentation to detect leakage within the requirements of Regulatory Guide 1.45 is actually improved. The containment atmosphere gaseous monitor is being deleted from TS, because, it does not meet the requirements of Regulatory Guide 1.45 to detect a 1.0 gpm [gallon per minute] RCS leakage within 1 hour. Extending the AOT [allowed outage time] when two of three leakage detection systems is inoperable does not decrease the margin of safety because one instrument remains operable, other instrumentation capable of indicating RCS leakage is available, and an RCS inventory balance is required to be performed on an increased frequency. The RCS inventory balance is more accurate than normal leak detection methods in regard to actual RCS leak rates, and therefore is an excellent alternative when other Ieak detection components may become inoperable. Maintaining diverse and accurate RCS leak detection methods available and capable of prompt leakage detection helps to ensure RCS leaks will be detected within an acceptable period of time and, therefore, the proposed changes do not significantly reduce the margin to safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 29, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.4.10, ''Reactor Coolant System—Structural Integrity, ASME Code Class 1, 2, and 3 Components,'' to relocate Surveillance Requirement (SR) 4.4.10.1.b which requires that the reactor vessel internals vent valves be tested and inspected, to the Technical Requirements Manual (TRM). The Davis-Besse Nuclear Power Station (DBNPS) TRM is a licensee-controlled document that is incorporated by reference into the DBNPS Updated Safety Analysis Report (USAR). Changes to the DBNPS TRM are performed in accordance with the regulatory requirements of 10 CFR 50.59.

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

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

Response: No. The proposed surveillance requirement relocation from the Technical Specifications to the USAR TRM does not alter the design, operation, or testing of any structure, system, or component. No preciously analyzed accident scenario is changed. Initiating conditions and assumptions remain as previously analyzed. Therefore, the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No. The proposed surveillance requirement relocation from the Technical Specifications to the USAR TRM does not alter the design, operation, or testing of any structure, system or component. The proposed change does not introduce any new or different accident initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No. The proposed surveillance requirements relocation from the Technical Specifications to the USAR TRM does not affect the capabilities of the Reactor Vessel Internals Vent Valves. Therefore, the proposed change will not affect a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 3, 2004.

Description of amendment request: The proposed amendment would change the facility as described in the Updated Safety Analysis Report (USAR) for the emergency diesel generators (EDGs). Specifically, the proposed change would describe a departure from Safety Guide 9, ''Selection of Diesel Generator Set Capacity for Standby Power Supplies,'' for the frequency and voltage transient during the EDG automatic loading sequence.

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

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

Response: No. The proposed amendment alters the design requirements for the Emergency Diesel Generators (EDGs). Specifically, the proposed amendment affects the requirements for EDG voltage and frequency response following a loss of offsite power. The EDGs function to mitigate the consequences of accidents when offsite power is not available. The EDGs are not an initiator of any analyzed accident.

The effect of this change on the capability of the EDGs, the onsite electric power system, and essentially powered equipment to perform their required safety functions has been evaluated, and the proposed change does not significantly impact the capability of these systems to perform their required accident mitigation functions. No previous analyzed accident scenario is affected by the proposed change.

The proposed change does not affect the initiation of any analyzed accident. The accident mitigation functions for affected equipment are maintained. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No. The proposed amendment affects the USAR requirements for EDG voltage and frequency response following a loss of offsite power. The effect of this change on the capability of the EDGs, the onsite electric power system, and essentially powered equipment to perform their required safety functions has been evaluated, and the proposed change does not significantly impact the capability of these systems to perform their required safety functions. The assumptions of the current accident analyses are maintained and no new or different accident initiators are created. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No. The proposed amendment affects the USAR requirements for EDG voltage and frequency response following a loss of offsite power. The effect of this change on the capability of the EDGs, the onsite electric power system, and essentially powered equipment to perform their required safety functions has been evaluated, and it is concluded the proposed change does not impact the capability of these systems to perform their required safety functions. However, since the proposed change does make changes to the controlling values for EDG voltage and frequency transient response that are less restrictive than those presently described in the USAR, this is considered a reduction in a margin of safety.

The magnitude of voltage and frequency drops which would result in failure of the EDGs, the onsite power system, or essentially powered equipment have not been determined due to the limitations of the transient assessment model and the nonlinear phenomena associated with that postulated failure. However, based on (1) a computer model and testing of the diesel engine, engine speed control governor and actuator, the synchronous generator and excitation system that demonstrate the EDGs are capable of starting, accelerating, and carrying the required loads, (2) a comprehensive evaluation of the impact of the transient voltage and frequency response on plant equipment and safety functions, (3) the momentary duration of the voltage and frequency dips, and (4) based on engineering judgement, the proposed change is not considered to have a significant effect on the margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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: April 23, 2004.

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

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. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes revise the Actions and allowed outage times of the accident monitoring instrumentation. The accident monitoring instrumentation is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased by these proposed changes. The Technical Specifications continue to require the accident monitoring instrumentation to be operable. Therefore, the accident monitoring instrumentation will continue to provide sufficient information on selected plant parameters to monitor and assess these variables following an accident. The consequences of an accident during the extended allowed outage time are the same as the consequences during the current allowed outage time. As a result, the consequences of any accident previously evaluated are not significantly increased by these proposed changes. Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different

kind of accident from any previously evaluated.

The proposed changes do not alter the design, physical configuration, or mode of operation of the plant. The accident monitoring instrumentation is not an initiator of any accident previously evaluated. No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed changes do not affect any other plant equipment. Therefore, operation of the facility in accordance with the proposed amendments does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes do not change the operation, function, or modes of the plant or equipment operation. The proposed changes do not change the level of assurance that the accident monitoring instrumentation will be available to perform its function. The proposed changes provide a more appropriate time to restore the inoperable channel(s) to operable status, and only apply when one or more channels of a required instrument are inoperable. The additional time to restore an inoperable channel to operable status is appropriate based on the low probability of an event requiring an accident monitoring instrument during the interval, providing a reasonable time for repair, and other means which may be available to obtain the required information. Therefore, operation of the facility in accordance with the proposed amendments would 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: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: William F. Burton, Acting.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: March 15, 2004.

Description of amendment request: Maine Yankee Atomic Power Company (Maine Yankee) is requesting that the U.S. Nuclear Regulatory Commission (NRC) release the remaining land under License No. DPR–36, with the exception of land where the Independent Spent Fuel Storage Installation is located. Maine Yankee submitted detailed information on dismantlement activities and final status survey results for the Spray Building and Spray Pipe with the amendment request, and proposes to submit dismantlement and survey information for the remaining land area in four additional submittals. Maine Yankee is seeking review and approval of the amendment; however, Maine Yankee is requesting that the NRC condition the effective date of the license amendment to correspond with the NRC's approval of the final information submittal.

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

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

Response: No.

The requested license amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. The release of this land will occur after all demolition activities are completed and final status surveys have been performed to document the final radiological conditions of the land. When the release occurs, the only remaining radiological hazard at the site will be contained in the Independent Spent Fuel Storage Installation (ISFSI). Therefore, the focus of the analysis is on the potential impact on the probability and consequences of accidents associated with the ISFSI.

The accident conditions evaluated for the spent fuel storage casks include the following: accident pressurization, mis-loading of fuel canisters, drop of the vertical concrete casks, explosion, fires, maximum anticipated heat load, earthquakes, floods, lightening strikes, tornado and tornado driven missiles, tip over of vertical concrete cask, and full blockage of vertical concrete cask air inlets and outlets. The release of the non-ISFSI land from the license will not affect the probability of any of these accidents. Maine Yankee will retain sufficient control over activities performed on the Owner Controlled Area through rights granted in the legal land conveyance documents to ensure that there is no impact on consequences from postulated accidents. Therefore, the proposed release of the land will not affect the consequences of any of these postulated accidents.

The proposed action, therefore, does not increase either the probability or the consequences of any accidents that have been considered.

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

Response: No.

The requested amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. When the amendment becomes effective, demolition activities will be complete and all systems, structures and components will have been removed from the land. The requested release of the land does not create the possibility of a new or different kind of accident that could affect the ISFSI that has not been considered in the design, installation or operation of the ISFSI. As noted above, Maine Yankee will retain control over activities performed in the Owner Controlled Area for the ISFSI to assure that no new hazards are introduced that could create the potential for a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use. This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Additionally, the State of Maine, through legislation, has imposed a 10 mrem/yr all pathways dose limit, with no more than 4 mrem/yr attributable to drinking water sources.

The License Termination Plan (LTP) prepared by Maine Yankee establishes conservative criteria for residual radiation levels following completion of demolition activities at the site. The LTP demonstrates that when these conservative criteria are met, the dose to the average member of the critical group will be below the regulatory criteria established by the State of Maine, and, therefore, well below the dose limits established by the NRC. The proposed release of the site lands, once the criteria established in the LTP have been met will, therefore, not result in any reduction in the margin of safety.

Conclusion

Based on the above, Maine Yankee concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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

Attorney for licensee: Joe Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578

NRC Section Chief: Claudia M. Craig.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: April 19, 2004.

Description of amendment request: The licensee proposed to revise the Technical Specifications (TSs) to establish an operating cycle (24-month) calibration surveillance frequency for the Intermediate Range Monitor (IRM) instrumentation, which would replace the current ''prior to startup and normal shutdown'' Surveillance Requirement (SR). The proposed changes also included associated conforming changes. In addition, the licensee proposed to relocate the Limiting Conditions for Operation (LCOs) and SRs for selected control rod withdrawal block instrumentation to the Updated Final Safety Analysis Report (UFSAR), a licensee-controlled document.

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

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

Response: No.

The proposed changes are limited to: (1) establishing a 24-month calibration frequency for the IRM instrumentation in lieu of the current ''prior to startup and normal shutdown'' requirement and incorporating the associated conforming changes, and (2) the relocation of certain instrumentation requirements from the TSs that do not satisfy the screening criteria for retention in the TSs. The proposed changes do not introduce any new modes of plant operation, make any physical changes to the plant, or alter any operational setpoints in a manner which could degrade the performance of, or increase the challenges to, any safety system assumed to function in the accident analysis. In addition, evaluations of the proposed changes pursuant to NRC and industry guidance demonstrate that the availability and reliability of equipment and systems required to prevent or mitigate the radiological consequences of an accident are not significantly affected. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes establish a 24-month IRM calibration frequency in lieu of the current ''prior to startup and normal shutdown'' requirement and relocate certain instrumentation requirements to the UFSAR. As such, the proposed changes do not eliminate any requirements or impose any new requirements, and adequate controls of existing requirements are maintained. Furthermore, since the proposed changes do not make any physical changes to the plant, no new accident initiators or failure mechanisms are introduced, and the accident assumptions and initial conditions will remain unchanged. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident [previously] evaluated.

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

Response: No.

The proposed changes establish a 24-month IRM calibration frequency in lieu of the current ''prior to startup and normal shutdown'' requirement and relocate certain instrumentation requirements to the UFSAR. Although the proposed changes result in changes to surveillance intervals, the impact, if any, on system availability is small based on (1) other more frequent testing that is performed, (2) the existence of redundant equipment, and (3) overall system reliability. Consistent with the findings of previous industry evaluations, the NMP1 [Nine Mile Point Nuclear Station, Unit No. 1] plant-specific analyses have shown no evidence of time-dependent failures that would impact the availability of the affected systems. Furthermore, plant-specific evaluations and the adoption of the calculated IRM setpoint Allowable Values ensure that the setpoint margins are maintained for a 24-month (30-month maximum) calibration frequency. The proposed relocated requirements are consistent with the Improved Standard TSs (NUREG–1433 and NUREG–1434) and 10 CFR 50.36, and will be maintained in accordance with 10 CFR 50.59. Accordingly, the proposed changes will have no significant impact on the condition or performance of structures, systems, and components relied upon for accident mitigation. 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment request: January 20, 2004.

Description of amendment request: This License Amendment Request (LAR) proposes selective scope application of the alternate source term (AST) for the fuel handling accident (FHA) in accordance with the provisions of 10 CFR 50.67. Nuclear Management Company requests the Nuclear Regulatory Commission (NRC) review and approval of the AST FHA methodology for application to the Prairie Island Nuclear Generating Plant. This LAR also proposes revisions to Technical Specifications (TS) associated with ensuring that safety analyses assumptions are met for a postulated FHA in containment. Based on the AST FHA analyses, this LAR proposes to modify TS 3.9.4, ''Containment Penetrations,'' to apply during the handling of recently irradiated fuel and require all containment penetrations to be closed during handling of recently irradiated fuel; and also proposes to remove the requirements of TS 3.3.5, ''Containment Ventilation Isolation Instrumentation'' relating to movement of irradiated fuel assemblies.

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

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

Response: No.

The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF–51, ''Revise containment requirements during handling irradiated fuel and core alterations' as it applies to TS 3.9.4, ''Containment Penetrations.'' This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, ''Containment Ventilation Isolation Instrumentation'' since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

This amendment does not alter the methodology or equipment used directly in fuel handling operations. None of the containment integrity features including the containment equipment hatch, personnel air locks or any other containment penetration are used to handle fuel. Therefore, containment integrity and ventilation systems, and spent fuel pool ventilation systems are not accident initiators and therefore these changes do not increase the probability of a previously evaluated accident.

The total effective dose equivalent (TEDE) doses from the analysis supporting this amendment request have been compared to equivalent total effective dose equivalent (TEDE) doses estimated with the guidelines of Regulatory Guide 1.183 Footnote 7. The new values are shown to be comparable to the results of the previous analysis.

A fuel handling accident analysis utilizing alternative source term methodologies allowed by 10 CFR 50.67 demonstrated that the dose consequences of a postulated fuel handling accident remain within the limits of 10 CFR 50.67 without taking credit for containment closure or ventilation systems assuming the fuel has not recently been in a critical reactor. The alternative source term fuel handling accident analysis also demonstrated that the more restrictive dose guidelines of Regulatory Guide 1.183 are also met without taking credit for these mitigation features. Since the alternative source term fuel handling accident analysis results are within the regulatory limits and regulatory guidelines without taking credit for these mitigation features, revising this Technical Specification for containment closure does not involve a significant increase in the consequences of a previously evaluated accident.

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

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

Response: No.

The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF–51, ''Revise containment requirements during handling irradiated fuel and core alterations'' as it applies to TS 3.9.4, ''Containment Penetrations.'' This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, ''Containment Ventilation Isolation Instrumentation'' since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

The proposed Technical Specification changes do not involve plant design, hardware, system operation, or procedures involved with actual handling of irradiated fuel. The proposed changes include application of new methodology for fuel handling accident analysis and revises requirements for equipment operability during movement of irradiated fuel assemblies. These changes do not create the possibility for 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. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

The proposed Technical Specification changes require containment integrity during movement of recently irradiated fuel. With this change, the Technical Specifications selectively implement 10 CFR 50.67 alternative source term methodologies for a fuel handling accident and implement portions of the approved industry improved Standard Technical Specification traveler, TSTF–51, ''Revise containment requirements during handling irradiated fuel and core alterations' as it applies to TS 3.9.4, ''Containment Penetrations.'' This change also removes requirements for containment ventilation isolation instrumentation during handling irradiated fuel from TS 3.3.5, ''Containment Ventilation Isolation Instrumentation'' since the containment purge and inservice purge system penetrations which are isolated by this instrumentation will be required to be isolated during movement of recently irradiated fuel. With the proposed 10 CFR 50.67 alternative source term methodologies, these filtration systems are not assumed to function during a fuel handling accident involving fuel which is not recently irradiated.

The assumptions and input used in the fuel handling accident analysis are conservative. The design basis fuel handling accident has been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to Regulatory Guide 1.183, Appendix B and with conservative assumptions concerning prior reactor operations. The control room atmospheric dispersion factor has been calculated with conservative assumptions associated with the release. These conservative assumptions and input ensure that the radiation doses cited in this license amendment request are the upper bounds to radiological consequences of a fuel handling accident in containment or the spent fuel pool. The analysis shows that there is a significant margin between the offsite radiation doses calculated for the postulated fuel handling accident using the alternate source term and the dose limits of 10 CFR 50.67 and acceptance criteria of Regulatory Guide 1.183. The proposed changes will not degrade the plant protective boundaries, will not cause a release of fission products to the public, and will not degrade the performance of any structures, systems, and components important to 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 requests involve no significant hazards consideration.

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

NRC Section Chief: L. Raghavan.

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.*

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

Date of application for amendments: April 17, 2003, as supplemented July 29, 2003.

Brief description of amendments: These amendments revise the Required Actions requiring suspension of operations involving positive reactivity additions and various notes that preclude reduction of boron concentration.

Date of issuance: May 6, 2004.

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

Amendment Nos.: 266 and 243.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 27, 2003 (68 FR 28841).

The July 29, 2003, letter clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on May 27, 2003 (68 FR 28841).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated May 6, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 5, 2003.

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

Date of issuance: April 29, 2004.

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

Amendment Nos.: 213, 207.

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

Date of initial notice in Federal Register: February 17, 2004 (69 FR 7520)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 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 5, 2003.

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

Date of issuance: April 29, 2004.

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

Amendment Nos.: 221, 203.

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

Date of initial notice in Federal Register: February 17, 2004 (69 FR 7520)

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 9, 2004, as supplemented by letter dated March 2, 2004.

Brief description of amendment: The amendment removed the pressurizer heatup and cooldown limits, and the associated action and surveillance requirements, from the Technical Specifications and placed them in the Technical Requirements Manual.

Date of issuance: May 4, 2004.

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

Amendment No.: 253.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 2, 2004 (69 FR 9860).

The March 2, 2004, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 4, 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, Units Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: July 18, 2002, as supplemented November 14, 2002, and December 11, 2003.

Brief description of amendments: The amendments relocate Technical Specification (TS) 3/4 9.7 regarding the Spent Fuel Storage Pool Building cranes and TS 3/4 9.13 (Unit 1) and TS 3/4 9.12 (Unit 2) regarding spent fuel cask cranes to the respective units' Updated Final Safety Analysis Report.

Date of Issuance: April 28, 2004

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

Amendment Nos.: 190 and 134

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 6, 2002 (67 FR 50954). The November 14, 2002, and December 11, 2003, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 2004.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: May 22, 2002, as supplemented by letters dated December 5, 2002, and February 11, 2004.

Brief description of amendment: The amendment revised Technical Specification 6.9.1.11.b to add two NRC-approved topical reports to the Core Operating Limits Report methodology list, and delete superseded reports. Also, the method of listing topical reports was revised to be consistent with Technical Specifications Task Force 363, which has been approved by the NRC.

Date of Issuance: May 6, 2004.

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

Amendment No.: 191.

Facility Operating License No. DPR–67: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 25, 2002 (67 FR 42827).

The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 6, 2004.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: August 22, 2003, as supplemented by letters dated January 12 and March 11, 2004.

Brief description of amendment: The amendment revised Section 3.7.1, ''Service Water (SW) System and Ultimate Heat Sink (UHS),'' by adding a new Condition G to allow continued operation with short-term elevated UHS temperatures.

Date of issuance: May 7, 2004.

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

Amendment No.: 113.

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

Date of initial notice in Federal Register: September 30, 2003 (68 FR 56344).

The January 12 and March 11, 2004, letters provided clarifying information within the scope of the original application, and did not change the staff's initial proposed no significant hazards consideration determination. The staff's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 30, 2004.

Brief description of amendment: The amendment relocates the requirements for hydrogen monitors from the Technical Specifications to the Technical Requirements Manual.

Date of issuance: May 13, 2004.

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

Amendment No.: 174.

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

Date of initial notice in Federal Register: March 2, 2004 (69 FR 9862).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 25, 2003, as supplemented on December 5, 2003

Brief description of amendment: The amendment modifies Technical Specification (TS) 2.1.4, ''Reactor Coolant System (RCS) Leakage Limits,'' by (1) adding a requirement for no RCS pressure boundary leakage, (2) combining the existing RCS leakage limits into a format similar to the Improved Standard TS (ISTS), and (3) replacing the existing basis associated with this TS with a basis similar in format and content to the ISTS.

Date of issuance: May 7, 2004.

Effective date: As of the date of issuance, to be implemented within 90 days from issuance.

Amendment No.: 226.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 19, 2003 (68 FR 49818).

The December 5, 2003, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 30, 2003, and its supplement dated March 11, 2004.

Brief description of amendments: The amendments eliminate the requirements in the technical specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: May 4, 2004.

Effective date: May 4, 2004, and shall be implemented within 60 days from the date of issuance.

Amendment Nos.: Unit 1—168; Unit 2—169.

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

Date of initial notice in Federal Register: March 2, 2004 (69 FR 9864).

The March 11, 2004, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 4, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 14th May 2004.

For the Nuclear Regulatory Commission.

Eric J. Leeds,

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

[FR Doc. 04–11507 Filed 5–24–04; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 52 Construction Inspection Program Framework Document; Availability of NUREG

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The Nuclear Regulatory Commission is announcing the completion and availability of NUREG–1789, ''10 CFR Part 52 Construction Inspection Program Framework Document,'' dated April 2004.

ADDRESSES: Copies of NUREG–1789 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402–9328; *http://www.access.gpo.gov/su_docs;* 202–512–1800 or The National Technical Information Service, Springfield, Virginia 22161–0002; *http://www.ntis.gov;* 1–800–533–6847 or, locally, 703–805–6000.

A copy of the document is also available for inspection and/or copying for a fee in the NRC Public Document Room, 11555 Rockville Pike, Rockville, Maryland. As of November 1, 1999, you may also electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at *http://www.nrc.gov/reading-rm.html.*