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

2 p.m. Briefing on Emergency Preparedness Program Initiatives (Closed—Ex. 1) (Contact: Nader Mamish (301) 415–1086).

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

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

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

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

Dated: November 4, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–25024 Filed 11–5–04; 9:34 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, October 15, 2004, through October 28, 2004. The last biweekly notice was published on October 26, 2004 (69 FR 62467).

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. (Note: Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) 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/.* (**Note:** Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) 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.* (**Note:** Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) 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.*

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

Date of amendment request: September 16, 2004.

Description of amendment request: The proposed amendment would revise the scope and the frequency of Surveillance Requirement (SR) 3.7.6.1 for verification of one complete cycle of each turbine bypass valve (TBV) every 92 days. The proposed change to SR 3.7.6.1 would allow a 5 percent stroke rather than a complete (100 percent) stroke of each TBV, and would extend the surveillance frequency from 92 days to 120 days. The complete stroke verification currently required by SR 3.7.6.1 once after each entry into MODE 4 would be retained and renumbered SR 3.7.6.2.

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

The proposed change to Technical Specification Surveillance Requirement (SR) 3.7.6.1 will allow a 5% stroke rather than a complete (100%) stroke of each turbine bypass valve (TBV), and will extend the surveillance frequency from 92 days to 120 days. The requirement to verify one complete cycle of each TBV once after each entry into MODE 4 will be retained.

The proposed testing requirements will provide a level of assurance, equivalent to that which now exists, that the TBVs will remain operable throughout the operating cycle, and that they will be able to perform their intended safety function if called upon to do so. Additionally, the reduction in the potential for plant transients that can result from the current testing requirements, will more than offset the small increase (less than one half of one percent) in TBV failure probability per cycle with the proposed testing regime. Thus the proposed changes will not significantly increase the probability of an accident previously evaluated.

Fermi 2 is analyzed for the increase in reactor pressure transient events with the assumption that the Main Turbine Bypass System (MTBS) is out-of-service. Feedwater Controller Failure Upscale represents the most limiting event in this analytical category, and provides the basis for the Minimum Critical Power Ratio (MCPR) operating limits that are applicable when the MTBS is out of service. Because the proposed testing requirements do not alter the assumptions for any of the increase in pressure transient events, the radiological consequences of an accident previously evaluated are not increased.

Therefore, this proposed amendment will not involve a significant increase in the probability or the 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 does not significantly affect the assumed performance of the TBVs, nor does it affect any other plant systems, structures, or components. In fact, these changes reduce the possibility of secondary plant transients and the potential for recirculation pump runbacks during the performance of this SR while at power. The proposed changes do not install any new plant equipment, nor is installed plant equipment being operated in a new or different manner. The proposed changes in test frequency and methodology will continue to ensure that the TBVs remain capable of performing their intended safety function. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will modify the scope and the frequency of the quarterly full stroke test of the TBVs. The operability requirements and functional characteristics of the TBVs remain unchanged. The proposed change to SR 3.7.6.1 from full stroke testing to 5% stroke testing, and from 92 days to 120 days has been evaluated to produce only a minimal increase in the failure probability of a TBV during each cycle (less than one half of one percent). This failure probability increase is outweighed by the reduction in the potential for plant transients resulting from full stroke testing during power operation. Both Alstom's sensitivity study, and actual industry experience at Ringhals Units 1 and 2 have shown that a partial stroke test will ensure that the valves remain mechanically operable throughout the operating cycle. The Alstom study further shows that a partial stroke test at 120 days, rather than at 92 days, will ensure that the valves remain mechanically operable throughout the operating cycle. Additionally, retaining the requirement to full stroke test each TBV once after each entry into MODE 4 will continue to verify that the valves are mechanically operable prior to their first use following each startup from MODE 4. The TBV response times are used in determining the effect on the MCPR. The surveillance test that ensures the MTBS meets the system's response time limits (SR 3.7.6.3) is not affected by these proposed changes and will continue to be performed at its current 18 month frequency. Therefore, this proposed change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: October 7, 2004.

Description of amendment request: The proposed amendment would revise the Safety Limit Minimum Critical Power Ratio in Technical Specification 2.1.1.2 to reflect the results of cycle-specific calculations performed for Fermi 2 operating Cycles 10 and 11.

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 basis of the Safety Limit Minimum Critical Power Ratio (SLMCPR) is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new CPR value preserves the existing margin to transition boiling and probability of fuel damage is not increased. The derivation of the revised SLMCPR for Fermi 2 for incorporation into the Technical Specifications, and its use to determine plant and cycle-specific thermal limits, have been performed using NRC approved methods. These plant-specific calculations are performed each operating cycle and if necessary, will require future changes to these values based upon revised core designs. The revised SLMCPR values do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

Therefore, this proposed amendment 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 results only from a specific analysis for the Fermi 2 Cycle 10 and 11 cores. This change does not involve any new or different methods for operating the facility. No new initiating events or transients result from these changes. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The new SLMCPR is calculated using NRC approved methods with plant and cycle-specific parameters for the Cycle 10 and 11 core designs. The SLMCPR value is established to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle-specific transients are evaluated. Accordingly, the margin of safety is maintained with the revised values. Therefore, this 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: August 18, 2004.

Description of amendment request: The proposed amendment would correct an inadvertent technical specification (TS) change associated with TS Amendment 184/166 and 182/164. Licensing Amendment 182/164 deleted the safety injection steam line pressure-low (SLPL) function and all concerned references due to redundant safety injection signals. This amendment was approved on September 22, 1998. As part of the conversion to standardized TS (STS), Amendment 184/166, all concerned references to the SLPL function were not correctly deleted from STS 3.3.2. Specifically, a reference to the SLPL function was not deleted from Footnote (c) to STS Table 3.3.2–1 and from the Basis of STS 3.3.2 Function 4.d.(1). Amendment (184/166) was approved on September 30, 1998.

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:

Criterion 1—Does This LAR Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

No. Approval and implementation of this LAR will have no effect on accident probabilities or consequences since the proposed changes are consistent with those previously reviewed and approved by the NRC in TS Amendment 182/164.

Criterion 2—Does This LAR Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

No. This LAR does not involve any physical changes to the plant. Therefore, no new accident causal mechanisms will be generated. The proposed changes are consistent with those previously reviewed and approved by the NRC in TS Amendment 182/164. Consequently, plant accident analyses will not be affected by these changes.

Criterion 3—Does This LAR Involve a Significant Reduction in a Margin of Safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be affected by the proposed changes since they are consistent with those previously reviewed and approved by the NRC in TS Amendment 182/164. Therefore, the proposed changes in this license amendment will not result in a significant reduction in the facility's margin of safety.

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

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

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: October 12, 2004.

Description of amendment request: The proposed license amendment request would change the Final Safety Analysis Report (FSAR) to reflect that the reactor core isolation cooling (RCIC) system is not required to mitigate the consequences of the control rod drop accident (CRDA). The FSAR revision would clarify that although the RCIC system is designed to initiate and inject into the reactor pressure vessel (RPV) at a low water level (L2), the additional RPV inventory is not required to prevent the accident or to mitigate the consequences of the CRDA.

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

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

This change clarifies, in various sections of the FSAR, that RCIC system operation is not required in order to mitigate the consequences of the CRDA. The proposed change involves no changes to plant systems or accident analyses. The accident analysis for the CRDA demonstrates that core design, the control rod pattern controls, and the scram signal from the reactor protection system (RPS) effectively prevent damage to the fuel rods as a result of the dropped rod. Furthermore, based on a prescribed source term provided from an assumed damage to less than 2% fuel in the core, the resulting radiological consequences are not affected by RCIC operation or failure to operate. As such, the change does not affect initiation of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change clarifies, in various sections of the FSAR, that the RCIC system operation is not required in order to mitigate the consequences of the CRDA. The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change clarifies, in various sections of the FSAR, that the RCIC system operation is not required in order to mitigate the consequences of the CRDA. The change has no effect on plant systems, operating practices or safety analyses assumptions. For these reasons, 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 30, 2004.

Description of amendment request: The proposed amendment would change the existing steam generator tube surveillance program to be consistent with that being proposed by the Technical Specifications Task Force (TSTF) in TSTF–449, Draft Revision 2. These proposed changes would revise the Technical Specifications and Bases for Specifications 3.4.13, RCS [Reactor Coolant System] Operational LEAKAGE, Specification 5.5.9, Steam Generator (SG) Tube Surveillance Program, and Specification 5.6.7, Steam Generator Tube Surveillance Reports, and add a new Specification 3.4.16 entitled Steam Generator (SG) Tube Integrity. Also, as a result of the licensee replacing the SGs with SGs having a new Alloy 690 thermally treated tubing design, the Technical Specifications and Bases would be revised to reflect this replacement.

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

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

Response: No.

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of design basis operating conditions (including startup, power operation, hot standby, cooldown, anticipated transients and postulated accidents). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. These criteria assure that the probability of an accident will not be increased.

The primary to secondary accident induced leakage rate for any design basis accidents, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. [The primary to secondary accident induced leakage rate is relatively inconsequential for the SG tube rupture analysis.] The operational LEAKAGE performance criterion meets current NRC regulations and NEI [Nuclear Energy Institute] 97–06 criteria for reactor coolant system (RCS) operational primary to secondary LEAKAGE through any one SG of 150 gallons per day. These criteria assure that accident doses will stay within regulatory and licensing basis limits.

Therefore, the proposed change does not affect the probability or consequences of any ANO–1 [Arkansas Nuclear One, Unit 1] analyzed accidents.

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

Response: No.

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. The proposed change enhances SG inspection requirements.

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

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

Response: No.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current technical specifications.

Therefore, the margin of safety is not changed by the proposed change to the ANO–1 TSs.

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

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

NRC Section Chief: Michael K. Webb, Acting.

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

Date of amendment request: September 30, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.2.1, Fuel Assemblies, to permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components. Also, the proposed amendment would modify TS 2.1.1.2, Reactor Core Safety Limits, to allow the use of the high thermal power (BHTP) correlation for departure from nucleate boiling (DNB) calculations of reload cores containing the Mark-B-HTP fuel design.

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 NRC approved topical reports BAW–10227P–A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR [Pressurized Water Reactor] Reactor Fuel, and BAW–10179P–A, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, provide the licensing basis for the Framatome ANP (FRA–ANP) advanced cladding and structural material, designated M5. The M5 material was shown in these documents to have equivalent or superior properties to the currently used Zircaloy-4 material. The cladding itself is not an accident initiator and does not affect accident probability. The M5 cladding has been shown to meet all 10 CFR 50.46 design criteria and, therefore, will not increase the consequences of an accident.

The proposed safety limit value ensures that fuel integrity will be maintained during normal operations and anticipated operational occurrences (AOOs), and that the design requirements will continue to be met. The core operating limits will be developed in accordance with the new methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

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.

Use of M5 clad fuel will not result in changes in the operation or configuration of the facility. Topical report BAW–10227P–A demonstrated that the material properties of the M5 alloy are similar or better than those of Zircaloy-4. Therefore, M5 fuel rod cladding and fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

In addition, there will be no change in the level of controls or methodology used for processing radioactive effluents or handling solid radioactive waste. Since the material properties of M5 alloy are similar or better than those of Zircaloy-4, there will be no significant changes in the types of any effluents that may be released off-site. There will not be a significant increase in occupational or public radiation exposure.

The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident / event initiator. No new initiating events or transients result from the use of the BHTP correlation or the related safety limit changes.

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

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

Response: No.

The proposed change will not involve a significant reduction in the margin of safety because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. M5 alloy is expected to perform similarly or better than Zircaloy-4 for all normal operating and accident scenarios, including both loss of coolant accident (LOCA) and non-LOCA scenarios. For LOCA scenarios, where the slight difference in M5 material properties relative to Zircaloy-4 could have some impact on the overall accident scenario, plant-specific LOCA analyses will be performed prior to the use of fuel assemblies with fuel rods or fuel assembly components containing M5. These LOCA analyses, required by the ANO–1 [Arkansas Nuclear One, Unit 1] TSs, will demonstrate that all applicable margins of safety will be maintained by the use of M5 alloy.

The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation, to ensure that the applicable margin of safety is maintained (*i.e.*, there is at least 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB). The other reactor core safety limits will continue to be met by analyzing the reload for the mixed core using NRC approved methods, and incorporation of resultant operating limits into the Core Operating Limits Report (COLR).

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

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

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

NRC Section Chief: Michael K. Webb, Acting.

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

Date of amendment request: September 1, 2004.

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: James W. Clifford.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota; Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa; Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin; Docket No. 50–255, Palisades Plant, Van Buren County, Michigan; Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin; Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment request: October 5, 2004.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) requirements for the licensee to submit annual occupational radiation exposure reports and monthly operating reports for the above nuclear plants. For the Kewaunee and Monticello plants, the licensee is also proposing to adopt a part of Revision 4 to TSTF–258, ''Changes to Section 5.0, Administrative Controls,'' regarding reporting challenges to, and failures, of certain safety/relief valves.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated October 5, 2004.

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

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

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

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

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

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: August 13, 2004.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool;'' TS 4.3.1.1, the criticality design features for fuel storage for VEGP Unit 1; and TS 4.3.1.2, the criticality design features for fuel storage for VEGP Unit 2. The proposed amendment would supplant the previous spent fuel rack criticality analysis with updated criticality calculations. Editorial revisions to TS Bases B 3.7.17, ''Fuel Storage Pool Boron Concentration,'' and B 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool,'' are included. In addition, Page vi of the Table of Contents will be updated to reflect the correct page number for Figure 5.5.6–1.

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

SNC has chosen to reanalyze the criticality analyses for the VEGP Unit 1 and Unit 2 spent fuel racks. Westinghouse performed the revised analyses using methods that address the non-conservatisms previously identified in the current analyses. The methodologies used for the revised analysis have been previously approved for use by the NRC.

The analyses revised the enrichment, burnup, and Integral Fuel Burnable Absorber (IFBA) limits required to comply with the allowed storage configurations. The storage configurations and interface requirements in the current Technical Specifications were retained in the revised analyses. The boron dilution evaluation that supported the initial amendments to permit credit for the soluble boron at VEGP continues to remain valid. The analyses demonstrated that Keff remains below unity for the various storage configurations considered with zero soluble boron and that Keff remains less than or equal to 0.95 for the entire pool with credit for soluble boron under non-accident and accident conditions with a 95% probability at a 95% confidence level (95/95).

Core design procedures ensure that new fuel can be stored in one or more of the allowed storage configurations. Administrative controls during fuel fabrication ensure that the fuel is fabricated accordingly to ensure proper loading of the fuel in the fuel assemblies. Administrative controls used to load fuel assemblies into the spent fuel pool ensure that fuel assemblies are stored in compliance with the allowed storage configurations. Fuel handling is performed under many administrative controls and physical limitations. These controls provide reasonable assurance that a criticality accident, fuel fabrication error, or fuel handling accident will not occur.

The change to the page number of Figure 5.5.6–1 on Page vi of the Table of Contents is administrative in nature.

Therefore, based on the conclusions of the above analysis, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The types of accidents previously evaluated include fuel fabrication errors, criticality accidents, and fuel handling accidents. The analyses revised the enrichment, burnup, and Integral Fuel Burnable Absorber (IFBA) limits required to comply with the allowed storage configurations. No new or other kind of accident can be postulated as a result of the revised analyses.

The change to the page number of Figure 5.5.6–1 on Page vi of the Table of Contents is administrative in nature.

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 decrease in the margin of safety?

The analyses revised the enrichment, burnup, and Integral Fuel Burnable Absorber (IFBA) limits required to comply with the allowed storage configurations. The boron dilution evaluation that supported the initial amendments to permit credit for soluble boron at VEGP was shown to remain valid. The analyses demonstrated that Keff remains below unity for the various storage configurations considered with zero soluble boron and that Keff remains less than or equal to 0.95 for the entire pool with credit for soluble boron under non-accident and accident conditions with a 95% probability at a 95% confidence level (95/95).

The change to the page number of Figure 5.5.6–1 on Page vi of the Table of Contents is administrative in nature.

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

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

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

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: July 8, 2004 (TS–427).

Description of amendment request: The proposed amendment removes the requirement to maintain an automatic transfer capability for the power supply to the Low Pressure Coolant Injection (LPCI) inboard injection and recirculation pump discharge valves. In addition, the licensee has requested to delete the references to Reactor Motor Operator Valve Boards D and E from Limiting Condition for Operation 3.8.7, and the Actions in 3.8.7 have been requested to be revised and/or renumbered, as appropriate.

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

Response: No.

Neither Reactor Motor Operated Valve (RMOV) Boards D and E, the equipment they power, nor the automatic power transfer feature provided for these boards are precursors to any accident previous [sic] evaluated in the Updated Final Safety Analysis Report (UFSAR). Therefore, the probability of an evaluated accident is not increased by modifying this equipment.

The proposed deletion of the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection and recirculation pump discharge valves does not change the number of Emergency Core Cooling System (ECCS) subsystems credited in the BFN licensing basis. Therefore, the proposed TS changes will not significantly increase the consequences of an accident previously evaluated.

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

Response: No.

The proposed deletion of the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection and recirculation pump discharge valves does not introduce new equipment, which could create a new or different kind of accident. No new external threats, release pathways, or equipment failure modes are created. Therefore, the proposed deletion of the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection and recirculation pump discharge valves will not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No.

The proposed deletion of the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection and recirculation pump discharge valves does not change the number of ECCS subsystems credited in the BFN licensing basis. The requirements of 10 CFR 50.46 and Appendix K continue to be met. Therefore, the proposed 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: August 2, 2004 (TS–435).

Description of amendment request: Modify the COMPLETION TIME for Technical Specification Limiting Condition for Operation (LCO) 3.6.3.1, Containment Atmosphere Dilution (CAD) System. The proposed change would extend the current completion time of 7 days with two CAD subsystems inoperable from existing requirement to shut down the reactor within 13 hours in accordance with LCO 3.0.3, when both CAD subsystems are inoperable.

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 safety-related function of the CAD system is to mitigate the effects of a loss-of-coolant-accident (LOCA) by limiting the volumetric concentration of oxygen in the primary containment atmosphere. The CAD System is not an event initiator, therefore, the probability of the occurrence of an accident is not affected by this proposed Technical Specification change. Emergency procedures preferentially use the normal containment inerting system to provide post accident vent and purge capability, with the CAD system only serving in a backup role to this system. Hence, in the event of the inoperability of both CAD subsystems, the proposed TS require the normal containment inerting system to be verified available as an alternate oxygen control means. Therefore, the proposed TS 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 introduce new equipment, which could create a new or different kind of accident. This proposed change does not result in any changes to the CAD equipment design or capabilities or to the operation of the plant. No new external threats, release pathways, or equipment failure modes are created. Therefore, the implementation of the proposed change will not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No.

As stated in GL [Generic Letter] 84–09, a Mark I type boiling water reactor (BWR) plant does not rely upon purge/repressurization systems such as CAD as its primary means of hydrogen control when the unit is operated in accordance with certain technical criteria. The BFN units are operated in accordance with these criteria. The BFN Unit 1 containment is inerted with nitrogen during normal operation, nitrogen from the containment inerting system with a backup from the CAD system is used for pneumatically operated components inside containment, and there are no potential sources of oxygen generation inside containment other than the radiolytic decomposition of water. The system preferred by the Emergency Operating Instructions (EOIs) for oxygen control post-accident is the normal primary containment inerting system. Because the probability of an accident involving hydrogen and oxygen production is small, CAD is not the primary system used to mitigate the creation of combustible containment atmosphere mixtures, and because the requested LCO where both CAD subsystems is inoperable is not long, no significant reduction in the margin of safety is associated with this proposed amendment.

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: Michael L. Marshall, Jr.

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

Date of amendment request: August 18, 2004.

Description of amendment request: The proposed amendment would update the reactor coolant system (RCS) and emergency core cooling system (ECCS) technical specifications (TSs). These changes include deleting TS 3/4.4.2, ''Safety Valves—Shutdown'' in its entirety, revising the action requirements for TS 3/4.4.3, ''Safety and Relief Valves—Operating,'' and deleting surveillance requirement 4.4.3.2.1.a for TS 3.4.3.2, ''Relief Valves—Operating.'' The proposed changes are consistent with the Sequoyah (SQN) safety analyses provided in the SQN Updated Final Safety Analyses Report and the improved standard technical specifications (NUREG–1431, Revision 3).

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. TVA's proposed TS revisions do not involve a significant increase in the probability of any accidents previously evaluated. TVA's proposed TS revisions provide improvements to the RCS and ECCS requirements to include appropriate reference to SQN's PTLR [PressureTemperature Limits Report] requirements. The proposed revision is a TS improvement that remains consistent with the improved standard TS requirements for Pressurized Water Reactors (PWRs) (NUREG–1431, Revision 3). TVA's proposed revision to delete SQN TS 3/4.4.2.1, ''Reactor Coolant Safety Valves—Shutdown,'' does not involve a significant increase in the probability of any accident previously evaluated. Pressurizer code safety valve requirements are not applicable for plant shutdown conditions (*i.e.,* modes 4 and 5) because the valves do not perform a safety function in these modes. The pressurizer code safety valves are not used as inputs to initiating events or accidents previously evaluated. Protection of the RCS against an overpressure condition in modes 4 and 5 is provided by the LTOP [low temperature overpressure protection] system which is governed by SQN TS 3.4.12. The setpoint for the pressurizer code safety valves is sufficiently high such that the safety valves do not afford protection to the RCS during low temperature operation. Accordingly, there is no impact on the consequences previously evaluated for the proposed change.

The proposed revisions are not the result of changes to plant equipment, test methods or operating practices. The proposed changes do not contribute to the generation or assumptions for postulated accidents. The proposed changes do not affect the design basis accidents or their assumptions. The revisions to SQN TSs continue to support SQN's required safety functions.

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 revisions are not the result of changes to plant equipment or plant design. The proposed revisions adopt standard TS requirements that are consistent with SQN's safety analysis and design and provide improvements over the existing requirements. The safety functions of the RCS and ECCS remain unchanged and do not affect any assumptions in SQN's accident analyses.

TVA's proposed change to delete the mode 4 and mode 5 TS requirements for pressurizer safety valves is consistent with the Policy Criterion of 10 CFR 50.36. The pressurizer code safety valves are not assumed to function for any safety analysis in modes 4 and 5 and consequently, the proposed changes do not create the possibility of a new or different kind of accident.

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

No. The proposed TS change does not involve a significant reduction in a margin of safety. TVA's proposed revisions will not result in changes to system design features or plant features that could be precursors to accidents or potential degradation of accident mitigation systems. The proposed changes to the RCS and ECCS requirements remain consistent with the current TS requirements for equipment operability. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

TVA's proposed change that removes the requirement for a pressurizer safety valve in modes 4 and 5 does not affect any margin of safety because the lift setting of the pressurizer code safety valves (2485 pounds per square inch gauge [psig] ±3 percent) is well above the limit needed to protect the RCS during low temperature operation and would not provide any safety function for overpressure protection in the lower modes. The TS requirements associated with low temperature operation are governed by SQN TS 3/4.4.12, LTOP system. The LTOP system provides the necessary overpressure protection for SQN's RCS in modes 4 and 5.

Accordingly, TVA's proposed deletion of operability requirements for SQN's pressurizer code safety valves for modes 4 and 5 will not affect the margin of safety.

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

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

NRC Section Chief: Michael L. Marshall, Jr.

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

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

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

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

Date of amendment request: October 12, 2004.

Brief description of amendment request: The proposed amendment would approve an engineering evaluation performed in accordance with Pilgrim Nuclear Power Station Technical Specification (TS) 3.6.D.3 to justify continued power operation with a safety relief valve discharge pipe temperature exceeding 212 degrees Fahrenheit for greater than 24 hours as required by TS 3.6.D.4.

Date of publication of individual notice in ***Federal Register***: October 20, 2004 (69 FR 61695).

Expiration date of individual notice: December 19, 2004.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: July 30, 2004.

Brief description of amendment: The proposed amendment would (1) add License Condition 2.C.(22) requiring an integrated tracer gas test of the control room envelope using methods described in American Society for Testing and Materials E741–00, ''Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution,'' and (2) delete Surveillance Requirement 3.7.3.6, which requires verification that unfiltered inleakage from control room emergency filtration system duct work outside the control room envelope is within limits.

Date of issuance: October 25, 2004.

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

Amendment No.: 162.

Facility Operating License No. NPF–43: Amendment adds a license condition and revises the Technical Specifications.

Date of initial notice in ***Federal Register***: August 13, 2004 (69 FR 50217)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2004.

No significant hazards consideration comments received: No.

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: October 21, 2003, as supplemented by letters dated February 10, 2004, and August 24, 2004.

Brief description of amendment: Modifies the Technical Specifications (TSs) to delete TS 3.6.4.4, ''Shield Building Annulus Mixing System'' and a reference to TS 3.6.4.4 within TS 3.10.1, ''Inservice Leak and Hydrostatic Testing Operation,'' and revise TS Surveillance Requirement 3.6.1.3.10, main steam isolation valve leakage limits.

Date of issuance: October 15, 2004.

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

Amendment No.: 143.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: May 25, 2004 (69 FR 29764). The supplement dated August 24, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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 16, 2004, as supplemented by letters dated June 8 and August 26, 2004.

Brief description of amendment: Modifies the Technical Specifications (TSs) to change Surveillance Requirement 3.6.5.1.3 of TS 3.6.5.1, ''Drywell,'' to allow a one-time extension of the test interval for the next drywell bypass leakage rate test from 10 years to 15 years.

Date of issuance: October 15, 2004.

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

Amendment No.: 144.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: May 25, 2004 (69 FR 29765). The supplements dated June 8 and August 26, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: January 29, 2004, as supplemented on April 12, June 16, June 30, July 16, August 3, August 12, and September 24, 2004.

Brief description of amendment: The amendment revises the operating license and Technical Specifications to authorize an increase in the maximum steady-state reactor core power level from 3114.4 megawatt thermal (MWt) to 3216 MWt. This represents a nominal increase of 3.26% rated thermal power.

Date of issuance: October 27, 2004.

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

Amendment No.: 241.

Facility Operating License No. DPR–26: Amendment revised the Facility Operating License and the Technical Specifications.

Date of initial notice in ***Federal Register***: March 2, 2004 (69 FR 9859). The April 12, June 16, July 16, August 3, August 12, and September 24, 2004, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

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: January 15, 2004, and supplemented on July 19, 2004.

Brief description of amendments: The amendments provide for an alternative means of testing the main steam Electromatic relief valves and the dual function Target Rock safety/relief valves.

Date of issuance: October 19, 2004.

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

Amendment Nos.: 211/203, 222/217.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: March 16, 2004.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 19, 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: August 19, 2003.

Brief description of amendments: The amendments modify Technical Specification (TS) 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to allow an exception to the testing guidance contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' Specifically, the TS change will allow potential valve atmospheric leakage paths (*e.g.*, valve stem packing) that are not exposed to test pressure during reverse-direction Type B or C tests (local leakage rate tests) to instead be tested during regularly scheduled Type A tests (integrated leakage rate tests).

Date of issuance: October 14, 2004.

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

Amendment Nos.: 168/154.

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

Date of initial notice in ***Federal Register***: December 23, 2003 (68 FR 74266).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 14, 2004.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of application for amendment: June 28, 2004, as supplemented September 3, 2004.

Brief description of amendment: The amendment revised the BVPS–1 Technical Specifications (TSs) surveillance requirements (SRs) 4.4.5.4.a.6, 4.4.5.4.a.8, and 4.4.5.5.d.1 and added SRs 4.4.5.4.a.11 and 4.4.5.5.e for Cycle 17 operation only. The change revised the definition of steam generator tube inspection scope in SR 4.4.5.4.a.8 to exclude the portion of the tube within the tubesheet below the W* distance, tube to tubesheet weld and tube-end extension by crediting the Westinghouse W* methodology as described in Topical Report WCAP–14797, Revision 2.

Date of issuance: October 15, 2004.

Effective date: This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

Amendment No.: 262.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 3, 2004 (69 FR 46584). The supplement dated September 3, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 9, 2003, as supplemented September 16, 2004.

Brief description of amendment: The amendment allows 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.

Date of issuance: October 21, 2004.

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

Amendment No.: 214.

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

Date of initial notice in ***Federal Register:*** March 30, 2004 (69 FR 16620). The September 16, 2004, supplemental letter provided additional information that clarified the application, but did not expand the scope of the application as originally noticed and did not change the U.S. Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: February 3, 2004.

Description of amendment request: This amendment revised a footnote to clarify a surveillance requirement and associated bases for emergency diesel generator testing.

Date of issuance: October 25, 2004.

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

Amendment No.: 98.

Facility Operating License No. NPF–86: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 16, 2004 (69 FR 12371).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 15, 2004, as supplement by letters dated September 28 and October 14, 2004.

Brief description of amendment: The amendment revises the Technical Specification (TS) Section 3.8.1, AC Sources—Operating, Condition B, to provide a one-time extension of the allowed outage time for one Diesel Generator (DG) inoperable from 7 days to 14 days and TS Section 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, Limiting Condition for Operation, to allow the use of temporary fuel oil storage tanks to supply the required fuel oil storage inventory.

Date of issuance: October 15, 2004.

Effective date: As of the date of issuance and shall be implemented on or before October 22, 2004.

Amendment No.: 207.

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

Date of initial notice in ***Federal Register***: August 3, 2004 (69 FR 46586). The supplements dated September 28 and October 14, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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 application for amendments: December 23, 2003, as supplemented by letter dated August 16, 2004.

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

Date of issuance: October 20, 2004.

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

Amendment Nos.: 167, 157.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: September 16, 2004 (69 FR 55844) The supplement dated August 16, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 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: December 1, 2003, as supplemented by letter dated July 2, 2004.

Brief description of amendment: The amendment changes the Fort Calhoun Station, Unit No. 1 Technical Specifications (TS) 2.7, ''Electrical Systems, TS Table 3–5, ''Minimum Frequencies for Equipment Tests,'' and TS 5.0, ''Administrative Controls,'' to modify the requirements for the diesel generator (DG) fuel oil for consistency with the Improved Standard Technical Specifications. The amendment also adds requirements for the DG lubricating oil and DG starting air.

Date of issuance: October 21, 2004.

Effective date: October 21, 2004, and shall be implemented within 120 days from the date of its issuance.

Amendment No.: 229.

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

Date of initial notice in ***Federal Register***: February 17, 2004 (69 FR 7526). The additional information provided in the supplemental letter dated July 2, 2004, did not expand the scope of the application as noticed and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated October 21, 2004.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: October 23, 2003, as supplemented by letters dated June 24, 2004 and August 26, 2004.

Brief description of amendment: The amendment revised Technical Specifications to delete the Surveillance Requirement associated with the emergency diesel generator lockout features.

Date of issuance: October 22, 2004.

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

Amendment No.: 155.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: December 9, 2003 (68 FR 68671). The June 24, 2004, and August 26, 2004, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original ***Federal Register*** notice.

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

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: December 12, 2003.

Brief description of amendment: The amendment revised the operating conditions for which Technical Specification (TS) 3/4.3.7.1, ''Radiation Monitoring Instrumentation,'' requires the control room ventilation radiation monitor to be operable. Additionally, the amendment revised the operating conditions for which TS 3/4.7.2, ''Control Room Emergency Filtration System,'' is applicable.

Date of issuance: October 28, 2004.

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

Amendment No.: 156.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** February 17, 2004 (69 FR 7527).

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

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: December 24, 2003, as supplemented by letter dated June 8, 2004.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to allow the use of GE14 fuel in reload cycle 13. Specifically, the change modified the TSs to reflect the use of General Electric (GE) core reload analysis methodology. The change revised the limiting conditions for operation for the recirculation loops to modify and add action statements to provide further thermal limit control during single-loop operation to be consistent with the GE methodology specified in the core operating limits report. The change also modified the TS definitions and TS requirements for average planar linear heat generation rate. Additionally, TS Section 6.9.1.9 is revised to correct an error from a previous amendment that inadvertently removed a reference.

Date of issuance: October 20, 2004.

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

Amendment No.: 154.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** February 17, 2004 (69 FR 7528). The June 8, 2004 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 2004.

No significant hazards consideration comments received: No.

Dated in Rockville, Maryland, this 1st day of November 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–24804 Filed 11–8–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Interim Staff Guidance Documents For Fuel Cycle Facilities

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

FOR FURTHER INFORMATION CONTACT: Wilkins Smith, Project Manager, Technical Support Group, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20005–0001. Telephone: (301) 415–5788; fax number: (301) 415–5370; e-mail: *wrs@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

The Nuclear Regulatory Commission (NRC) plans to issue Interim Staff Guidance (ISG) documents for fuel cycle facilities. These ISG documents provide clarifying guidance to the NRC staff when reviewing either a license application or a license amendment request for a fuel cycle facility under 10 CFR part 70. The NRC is soliciting public comments on the ISG documents which will be considered in the final versions or subsequent revisions.

II. Summary

The purpose of this notice is to provide the public an opportunity to review and comment on a draft Interim Staff Guidance document for fuel cycle facilities. Interim Staff Guidance–09 provides guidance to NRC staff relative the requirements associated with the use of Initiating Event Frequencies (IEFs) for demonstrating compliance with the performance requirements of 10 CFR 70.61.

III. Interim Staff Guidance–09, Initiating Event Frequency, Draft October 20, 2004 Issue

This guidance addresses the measures needed to assure the validity and maintenance of initiating event frequencies (IEFs) used to demonstrate compliance with the performance requirements for 10 CFR 70.61.

Introduction

The purpose of this Interim Staff Guidance (ISG) is to clarify the use of IEFs for demonstrating compliance with the performance requirements of 10 CFR 70.61. NUREG–1718, ''Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,'' and NUREG–1520, ''Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,'' provide methods for reviewing integrated safety analyses (ISAs), employing a semi-quantitative risk index method. While one of these methods is used below to illustrate the use of IEFs, applicants and licensees may use other methods which would produce similar results. There is no particular method explicitly mandated, and sequences that are risk significant or marginally acceptable are candidates for more detailed evaluation by the applicant or licensee and reviewer.

Discussion

Each licensee or applicant is required to perform an ISA to identify all credible high-consequence and intermediate-consequence events. The risk of each such credible event is to be limited through the use of appropriate engineered and/or administrative controls to meet the performance requirements of 10 CFR 70.61. Such a control is referred to as an item relied on for safety (IROFS). In turn, a safety program must be established and maintained to assure that each IROFS is available and reliable to perform its intended function when needed. The safety program may be graded such that management measures applied are graded commensurate with the reduction of risk attributable to that item. In addition, a configuration management system must be established pursuant to § 70.72, to evaluate changes, to assure, in part, that the IROFS are not removed without at least equivalent replacement of the safety function.

The risk of each credible event is determined by cross-referencing the severity of the consequence of the unmitigated accident sequence with the likelihood of occurrence in a risk matrix with risk index values. The likelihood of occurrence risk index values can be determined by considering the criteria in NUREG–1520, Tables A–9 through A–11. Accident sequences result from initiating events which are followed by the failure of one or more IROFS. Initiating events can be (1) an external event such as a hurricane or earthquake, (2) a facility event external to the process being analyzed (*e.g.*, fires, explosions, failures of other equipment, flooding from facility water sources), (3) deviations from normal operations of the process (credible abnormal events), or (4) failures of an IROFS in the process. Additional guidance regarding initiating probabilities from natural phenomena hazards are addressed in ISG–08, *Natural Phenomena Hazards.*

An initiating event does not have to be an IROFS failure. An item only becomes an IROFS if it is credited in the ISA for mitigation or prevention per the definition in § 70.4. If an item, whose failure initiates an event, has strictly an operational function, it does not have to be an IROFS. This applies to external events and can apply to internal events. If the item whose failure initiates an event, has solely a safety function that is credited in the ISA, then it should be an IROFS. If the item has both an operational and a safety function, the safety function should make it an IROFS (for its ISA credited safety features only).

IEFs can play a significant role in determining whether the performance requirements of § 70.61 are met for a particular accident sequence. Whether an initiating event is due to an IROFS or a non-IROFS failure, licensees should take appropriate action to assure that any change to the basis for assigning an IEF value to that event is evaluated on a continuing basis to ensure continued compliance with the performance requirements. For example, a non-IROFS component may not be subject to the same QA program controls and other management measures that an IROFS