TABLE 2.—SUMMARY OF RADIOLOGICAL ENVIRONMENTAL IMPACTS

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|---|---|
| Radiological Waste Stream | No change in design or operation of waste streams. |
| Gaseous Waste | Slight increase in amount of radioactive material in gaseous effluents; within FES estimate; offsite doses would continue to be well within NRC criteria. |
| Liquid Waste | Slight increase in amount of radioactive material in liquid effluents; within FES estimate; offsite doses would continue to be well within NRC criteria. |
| Solid Waste | No significant change in radioactive resins; no significant changes in dry waste; no significant changes in irradiated components. |
| Dose Impacts Occupational Dose | Up to 9.6 percent increase in collective occupational dose possible; well within FES estimate. |
| Offsite Direct Dose | Slight increase possible; not significant; offsite doses would continue to be within NRC criteria. |
| Postulated Accidents | Up to 9.6 percent increase in calculated doses from some postulated accidents; calculated doses within NRC criteria. |
| Fuel Cycle and Transportation | Increase in bundle average enrichment. Fuel enrichment and burnup would continue to be within bounding assumptions for Tables S–3 and S–4 in 10 CFR Part 51, ''Environmental Protection Regulations for Domestic Licensing and Related Regulatory Function;'' conclusions of tables regarding impact would remain valid. |

Alternatives to Proposed Action

As an alternative to the proposed action, the NRC staff considered denial of the proposed EPU (*i.e.*, the ''no-action alternative''). Denial of the application would result in no change in the current environmental impacts; however, other fossil-fuel generating facilities may need to be built in order to maintain sufficient power-generating capacity. As an alternative, the licensee could purchase power from power generating facilities outside the service area. The additional power would likely also be generated by fossil fuel facilities. Construction and operation of a fossil-fueled plant would create impacts in air quality, land use, and waste management significantly greater than those identified for the EPU at Waterford 3.

Implementation of the proposed EPU would have less impact on the environment than the construction and operation of a new fossil-fueled generating facility or the operator of fossil facilities outside the service area. Furthermore, the EPU does not involve environmental impacts that are significantly different from those presented in the 1981 FES for Waterford 3.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the 1981 FES for Waterford 3.

Agencies and Persons Consulted

In accordance with its stated policy, on August 13, 2004, the NRC staff consulted with the Louisiana State official, Ms. Nan Calhoun of the LDEQ, regarding the environmental impact of the proposed action. The State official had no comments.

**Finding of No Significant Impact**

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

**DATES:** The comment period expires November 12, 2004. Comments received after this date will be considered if it is practical to do so, but the Commission is able to assure consideration of comments received on or before this date.

**ADDRESSES:** Submit written comments to Chief, Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Mail Stop T–6 D59, Washington, DC 20555–0001. Written comments may also be delivered to 11545 Rockville Pike, Room T–6D59, Rockville, Maryland 20852, from 7:30 a.m. to 4:15 p.m. on Federal workdays. Copies of written comments received will be electronically available at the NRC's Public Electronic Reading Room link *http://www.nrc.gov/reading-rm/adams.html* on the NRC Homepage or at the NRC's Public Document Room located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. 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 Reference staff at 1–800–397–4209, or 301–415–4737, or by e-mail at *pdr@nrc.gov.*

**SUPPLEMENTARY INFORMATION:** The NRC is considering issuance of an amendment to Facility Operating License No. NPF–38 issued to Entergy for operation of Waterford 3 located in St. Charles Parish, Louisiana.

**FOR FURTHER INFORMATION CONTACT:** N. Kalyanam, Office of Nuclear Reactor Regulation, Mail Stop O–7D1, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, by telephone at (301) 415–1480, or by e-mail at *nxk@nrc.gov*.

Dated at Rockville, Maryland, this 30th day of September 2004.

For the Nuclear Regulatory Commission.

**Michael K. Webb,**

*Acting Chief, Section 1, Project Directorate IV, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.*

[FR Doc. 04–22786 Filed 10–8–04; 8:45 am]

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## 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 September 17, 2004, through September 30, 2004. The last biweekly notice was published on September 28, 2004 (69 FR 57978).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

*Date of amendment request:* April 14, 2004.

*Description of amendment request:* The proposed amendment would make changes to the Technical Specifications (TSs) that will eliminate secondary containment operability requirements when handling sufficiently decayed irradiated fuel and performing core alterations, and will clarify requirements associated with operations with potential to drain the reactor vessel. This proposed amendment also uses Alternate Source Term (AST) methodology in accordance with 10 CFR 50.67 for calculating Fuel Handling Accident (FHA) consequences. The proposed amendment also removes TSs operability requirements for engineered safety features (ESF) (*e.g.* primary/secondary containment, standby gas treatment, and isolation capability) after the sufficient decay of ''recently'' irradiated fuel.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

1. Do the Proposed Changes Involve a Significant Increase in the Probability or Consequence of an Accident Previously Evaluated?

The proposed changes do not modify the design or operation of equipment used to handle and move new and spent fuel or to perform core alterations. The proposed amendment does not modify the design of the ESF equipment. The proposed changes, therefore, will not increase the probability of accidents previously evaluated.

AST analysis does not affect the performance of the systems or components used to mitigate the consequences of accidents previously evaluated. While a direct comparison between current methodologies used in the current Pilgrim design basis analysis and Regulatory Guide (RG) 1.183 is not possible due to different acceptance criteria, the AST calculations demonstrate that the radiological consequences to the accidents previously evaluated will still remain below the regulatory limits. Therefore, any potential change in the radiological consequences are not considered significant. Since the radiological consequences are below the regulatory limits and the probability of an accident is unchanged, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the Proposed Changes Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Analyzed?

There are no new plant operation modes or physical modifications being proposed. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Do the Proposed Changes Involve a Significant Reduction in the Margin of Safety?

The licensee performed a comprehensive analysis and evaluation of the FHA using AST methodology and dose consequence analysis in accordance with 10 CFR 50.67. While direct comparison between methodologies used in the current Pilgrim design basis analysis and RG 1.183 is not possible due to different acceptance criteria, the revised doses will, however, remain below the total effective dose equivalent dose regulatory limits for the control room, exclusion area boundary, and low population zone as specified in 10 CFR 50.67. Therefore, by meeting the applicatory regulatory limits for AST, any potential decrease in a margin of safety would not be considered significant. The changes are, therefore, not considered a significant reduction in a margin of safety.

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

*NRC Acting Section Chief:* Daniel S. Collins.

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

*Date of amendment request*: June 17, 2004.

*Description of amendment request*: The proposed amendment would delete entries from Technical Specification (TS) Tables 3.2.6 and 4.2.6 related to the post-accident hydrogen and oxygen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Combustible gas control system for nuclear power reactors,'' eliminated the requirements for hydrogen recombiners (not installed at Vermont Yankee and therefore not addressed by this proposed amendment) and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated June 17, 2004.

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

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

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

With the elimination of the design-basis LOCA hydrogen release, hydrogen and oxygen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen and oxygen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, classification of the oxygen monitors as Category 2, and removal of the hydrogen and oxygen monitors from TSs will not prevent an accident management strategy through the use of the severe accident management guidelines, the emergency plan, the emergency operating procedures, and site survey monitoring that support modification of emergency plan protective action recommendations.

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

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

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

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

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

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

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

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

Category 2 Oxygen Monitors Are Adequate To Verify the Status of an Inerted Containment.

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

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

*Attorney for licensee*: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

*NRC Section Chief:* Allen G. Howe.

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

*Date of amendment request*: September 1, 2004.

*Description of amendment request*: The proposed amendment would delete Technical Specification (TS) 6.6.A, ''Occupational Radiation Exposure Report,'' and TS 6.6.B, ''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:

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

*Response*: No.

The proposed change eliminates the 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 TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

*Response:* No.

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

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

*Response:* No.

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

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

*Attorney for licensee:* Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

*NRC Section Chief:* Allen G. Howe.

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

*Date of amendment request:* April 30, 2004.

*Description of amendment request:* The proposed change allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions are revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement (SR) 3.0.4 is revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated April 30, 2004.

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

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

*NRC Section Chief:* Daniel Collins, Acting.

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

*Date of amendment request:* August 2, 2004.

*Description of amendment request:* The proposed amendment would delete Technical Specification (TS) Section 6.8.4, ''Post Accident Sampling,'' and the related requirements to maintain a Post-Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the U.S. Nuclear Regulatory Commission's (NRC) lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 3, 2003 (68 FR 10052) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in a license amendment application in the **Federal Register** on May 13, 2003 (68 FR 25664). The licensee affirmed the applicability of the following NSHC determination in its application dated August 2, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

*NRC Section Chief:* Anthony J. Mendiola.

*Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan*

*Date of amendment requests:* April 13, 2004.

*Description of amendment requests*: The proposed amendments would change the licensing basis as described in the Updated Final Safety Analysis Report to allow the use of a reinforcing bar (rebar) yield strength value based on measured material properties, as documented in the licensee rebar acceptance tests, in control rod drive missile shield structural calculations.

*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 of occurrence or consequences of an accident previously evaluated?

*Response:* No

Probability of Occurrence of an Accident Previously Evaluated

This is a change in the method of determining the acceptability of accommodating the pressure load following a loss-of-coolant accident. No physical changes are being made to the plant and no potential accident initiators are introduced by this change. Thus, the probability of the occurrence of any accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated

There is reasonable assurance that the ability of control rod drive missile shields (missile shields) to maintain their structural capability and continue to function as a part of the divider barrier separating the lower containment from the upper containment is not impacted by this change. The data obtained from rebar acceptance test reports demonstrate that the missile shields have adequate strength to accommodate the load that would be imposed under assumed accident conditions. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

*Response:* No

The use of increased missile shield rebar yield strength for the missile shield structural capability under accident conditions does not alter the evaluation of the missile shields' structural capability during normal operation, the operational condition in which a new or different kind of accident would be initiated. The change does not physically alter plant components nor does it alter plant operation. The change does not adversely affect current system interfaces or create new interfaces that could result in an accident or malfunction of a different kind than previously evaluated.

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

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

*Response:* No

The margin of safety for the missile shields is provided by the factors that are applied to the individual loads determining the load imposed on the missile shields under accident conditions. These code safety factors are sufficient to ensure that both anticipated and unanticipated loads can be withstood by the concrete structures. The use of yield strengths based on measured material properties as documented in the I&M [Indiana Michigan Power Company] rebar acceptance tests for the missile shield structural evaluation has no effect on the margin of safety provided by the load safety factors. I&M continues to use the same load factors that were used to license the Donald C. Cook Nuclear Plant.

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 requests involve no significant hazards consideration.

*Attorney for licensee*: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

*NRC Section Chief:* L. Raghavan.

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

*Date of amendment request*: September 7, 2004.

*Description of amendment request*: The proposed amendment revises Fort Calhoun Station (FCS) Technical Specification (TS) 5.9.5, ''Core Operating Limits Report,'' such that it will read consistent with TS 5.6.5 of NUREG–1432, Standard Technical Specifications-Combustion Engineering Plants. In addition, the list of core reload analysis methodologies contained in TS 5.9.5b used to determine the core operating limits is updated to move many of these references to Omaha Public Power District (OPPD) core reload analysis methodology documents OPPD–NA–8301, 8302, and 8303. Several analytical method references that are no longer applicable to FCS are deleted from TS 5.9.5b; several references will remain, as they are not suitable for incorporation into the core reload analysis documents.

*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 changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are primarily administrative in nature to achieve consistency with Standard Technical Specifications and to update the list of NRC reviewed and approved analytical methods used to develop core operating limits. Several of the analytical methods are no longer applicable to Fort Calhoun Station, Unit No. 1, (FCS) and thus are deleted from the Technical Specifications (TSs). Many of the topical reports currently referenced in TS 5.9.5b are more suitably referenced in the OPPD core reload methodology documents where they have been relocated.

The OPPD core reload methodology documents remain referenced in TS 5.9.5b and as such are subject to NRC review and approval. The relocation of the topical reports referenced in TS 5.9.5b to OPPD core reload methodology documents is an administrative change. In addition to the incorporation of references currently found in TS 5.9.5b, OPPD core reload methodology documents OPPD–NA–8301, 8302, and 8303 are revised to remove characters designating them as proprietary, and approved. This is an administrative change, as OPPD no longer considers the documents to be proprietary or topical reports. OPPD core reload methodology documents OPPD–NA–8301, 8302, and 8303 are enclosed for NRC review and approval [attached to the licensee's September 7, 2004, letter] of the changes noted above and incorporation of the CASMO–4 (C–4) computer code, which is described below.

OPPD is adding the C–4 code to OPPD–NA–8302, Reload Core Analysis Methodology, Neutronics Design Methods and Verification and will use the code for nuclear design analysis. This will allow the use of the C–4 and SIMULATE–3 (S–3) methodology to perform all steady-state pressurized water reactor (PWR) core physics analyses. The probability of occurrence of an accident previously evaluated will not be increased by the proposed change in the particular codes used for physics calculations for nuclear design analysis. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the Updated Safety Analysis Report (USAR). These inputs do not alter the physical characteristics or modes of operation of any system, structure, or component involved in the initiation of an accident. Thus, there is no significant increase in the probability of an accident previously evaluated as a result of this change.

The consequences of an accident evaluated in the USAR are affected by the value of inputs to the transient safety analysis. An extensive benchmark of C–4/S–3 predictions was performed with measured data using a variety of fuel designs and operating conditions in power reactors and critical experiments. The accuracy of C–4/S–3 is similar to, and sometimes better than, the accuracy of C–3/S–3. Furthermore, there is always the potential for the value of the nuclear design parameters to change solely as a result of the new core reload fuel core loading pattern. Regardless of the source of a change, an assessment is always made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents previously evaluated in the USAR. Refueling is an anticipated activity, which is described in the USAR. If increased consequences are anticipated, compensatory actions are implemented to neutralize any expected increase in consequences. These compensatory actions include, but are not limited to, crediting any existing margins in the analysis or redefining the operating envelope to avoid increased consequences. Thus, the nuclear design parameters are intermediate results and by themselves will not result in an increase in the consequence of an accident evaluated in the USAR.

Therefore, the use of the C–4/S–3 code package, which will perform the same functions as the C–3/S–3 codes with similar accuracy, does not significantly increase 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 create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes are primarily administrative in nature. The changes achieve consistency with Standard Technical Specifications, update the list of NRC reviewed and approved analytical methods used to develop core operating limits by deleting certain analytical methods no longer applicable to FCS and relocating many of the remainder to OPPD core reload analysis methodology documents, and make minor administrative changes to OPPD core reload analysis documents referenced in TS 5.9.5b. OPPD intends to utilize the C–4/S–3 code package for nuclear design analysis. The proposed amendment would add the C–4 code to OPPD core reload analysis methodology document OPPD–NA–8302.

The possibility for a new or different kind of accident evaluated previously in the USAR will not be created by the proposed administrative changes or the change to the particular codes used for physics calculations for nuclear design analyses. The change involves adding the Studsvik C–4 code to OPPD core reload analysis methodology document OPPD–NA–8302. The C–4 code is an update to the C–3 code currently approved for use at FCS. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the USAR. These inputs do not alter the physical characteristics or modes of operation of any system, structure or component involved in the initiation of an accident. Therefore, these administrative changes and the addition of the C–4 code, which will perform the same functions, as the C–3 code with similar accuracy, does not increase 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.

The proposed change does not involve a significant reduction in a margin of safety. The margin of safety as defined in the basis for any technical specification will not be reduced nor increased by the proposed administrative changes or the change to the codes used for physics calculations for nuclear design analyses. The changes achieve consistency with Standard Technical Specifications, update the list of NRC approved analytical methods used to develop core operating limits by deleting certain analytical methods no longer applicable to FCS and relocating many of the remainder to OPPD core reload analysis methodology documents, and make minor administrative changes to OPPD core reload analysis documents referenced in TS 5.9.5b.

The change involves the addition of the Studsvik C–4 code to OPPD core reload analysis methodologies for nuclear design analysis. Extensive benchmarking of the C–4/S–3 computer codes has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained using the NRC approved C–3/S–3 computer codes. For any changes in the calculated values that do occur, the application of appropriate biases and uncertainties ensures that the current margin of safety is maintained. Specifically, use of these code specific biases and uncertainties in safety evaluations continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis are conservative with respect to the actual values on at least a 95/95 probability/confidence basis.

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*: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

*NRC Section Chief:* Robert Gramm.

*PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey*

*Date of amendment request*: April 15, 2004, as supplemented August 11, 2004.

*Description of amendment request*: The amendment request proposes to revise the Salem Unit No. 1 Technical Specifications (TSs) to reflect the addition of the chilled water system to provide cooling water to the containment fan cooling units (CFCUs). The amendment request also proposes to revise a non-conservative Action Statement for Salem Unit Nos. 1 and 2 that allows three containment cooling fans to be inoperable under certain conditions.

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

*Response:* No.

Containment cooling fans remove containment heat loads under both normal and accident conditions. As such, they have no impact on the probability of occurrence of any previously evaluated accidents, although they do function to mitigate accident consequences. With regard to accident consequences, revised containment response analysis has been performed with the proposed changes of this license amendment. This analysis demonstrates that containment pressure and temperature limits continue to be met as further described below.

The addition of the non-safety related chilled water system does not represent an increase in the consequences of an accident since, at the onset of the accident, the chilled water supply is automatically isolated on the resulting safety injection signal and the safety related Service Water System supplies the cooling method to remove the containment heat loads, as presently analyzed. Analysis has been performed to evaluate any potential failures that could prevent the Containment Cooling System to perform [sic] its safety related functions. Redundancy in the chilled water system and transfer to service water during an accident are incorporated in the design. In addition, as a conservative measure, an action statement has been added to require prompt action to restore containment cooling or commence a unit shutdown in the event of an unexpected condition that results in the loss of normal containment cooling capability.

The accidents previously evaluated that are associated with containment heat removal are design basis loss-of-coolant accident (LOCA) and main steam line break (MSLB) accident. In the case of the design basis LOCA, the revised analysis demonstrates that all cases resulted in a peak containment pressure that was less than 47 psig. In addition, all long-term cases were well below 50% of the peak value within 24 hours. Based on the results, applicable criteria for Salem Unit 1 have been met and therefore, the consequences of previously evaluated accidents are not increased.

The proposed change to the non-conservative TS 3.6.2.3 Action b, maintains that five CFCUs remain operable to ensure that, upon a single failure, a minimum of three CFCUs will provide the required containment and air mixing which is consistent with the current Salem Dose Analysis.

Consequently, the proposed license amendment does not increase the probability of occurrence or the consequences of accidents previously evaluated for Salem.

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

*Response:* No.

Containment cooling fans remove containment heat loads under both normal and accident conditions. The containment cooling fans are presently part of the plant protection equipment and have been analyzed and evaluated as to their function and effectiveness. Consequently, they cannot create the possibility of any new or different kinds of accidents from any previously evaluated. The addition of a chilled water system that is isolated on an accident condition does not create a new or different kind of accident. The accidents analyzed are the LOCA and MSLB, which are part of the Salem Design Bases.

The proposed change to the non-conservative TS 3.6.2.3 Action b, maintains that five CFCUs remain operable to ensure that, upon a single failure, a minimum of three CFCUs will provide the required containment and air mixing which is consistent with the current Salem Dose Analysis.

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

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

*Response*: No.

The margin of safety pertinent to the proposed changes is the dose consequences resulting from a design basis LOCA. Containment cooling fans affect potential dose consequences in that they assist in maintaining containment pressure and temperature within design limits. By maintaining these limits, three critical functions are performed. These are:

a. Containment integrity is assured by maintaining pressure below the containment design limit.

b. By maintaining pressure below 47 psig, leakage of containment atmosphere to the surrounding environment is retained within the leakage testing results of 10 CFR 50, Appendix J. In this case, the Appendix J testing procedures provide the margin of safety, as long as the limiting pressure (47 psig) is not exceeded.

c. By maintaining containment temperature within limits, the qualification of vital electrical equipment to function in the post-accident containment environment is assured. In this case, the margin of safety is provided by the testing and evaluation procedures implemented by 10 CFR 50.49.

In addition, as a conservative measure, an action statement has been added to require prompt action to restore containment cooling or commence a unit shutdown in the event of an unexpected condition that results in the loss of normal containment cooling capability.

The proposed change to the non-conservative TS 3.6.2.3 Action b, maintains that five CFCUs remain operable to ensure that, upon a single failure, a minimum of three CFCUs will provide the required containment and air mixing which is consistent with the current Salem Dose Analysis.

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*: Jeffrie J. Keenan, Esquire, Nuclear Business Unit–N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

*NRC Section Chief:* James W. Clifford.

*R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York*

*Date of amendment request*: July 26, 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 July 26, 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 NRC staff proposes to determine that the requested change does not involve significance hazards consideration.

*Attorney for licensee*: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

*NRC Section Chief:* Richard J. Laufer.

*South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1, Fairfield County, South Carolina*

*Date of amendment request*: August 24, 2004.

*Description of amendment request*: The proposed change will revise Surveillance Requirements (SRs) 4.7. 1.2.a.1 and 4.7 .1.2.a.2 to reflect a more representative model of the Emergency Feedwater (EFW) System. The new model has established new technical specification (TS) acceptance criteria to assure the design requirements of the system are met. These required characteristics are more stringent than those currently in the VCSNS TSs for this system.

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

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

This change represents a more restrictive surveillance requirement than currently exists for TS Surveillance 4.7. 1.2.a.1 and 4.7 .1.2.a.2. These proposed surveillance acceptance criteria changes will ensure that the motor driven EFW pumps and the turbine driven EFW pump can continue to perform their design function. There are no changes planned to any plant installed hardware or software and normal plant operations will not be impacted.

The probability or consequences of accidents previously evaluated in the VCSNS FSAR [Final Safety Analysis Report] are unaffected by this proposed change because there is no change to any equipment response or accident mitigation scenario. There are no additional challenges to fission product barrier integrity. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change involves the revision of the Surveillance Requirements for the EFW system. The revised requirements are more restrictive to insure compliance with the design basis of the system. Changes to the system model require changes to the SR acceptance criteria in order to maintain the performance level assumed in the safety analysis.

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

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

The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. A change to the SR is proposed, however, the proposed change is more restrictive than the current SR. The more restrictive criteria inherently include a 5 gpm leak tolerance for the EFW flow control valves. This represents a built in margin for the pump head requirement when the flow control valve leakage is determined to be less than 5 gpm. Therefore, the proposed change does not involve a significant reduction in 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 G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

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

*Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia; Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama; Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia*

*Date of amendment request*: July 28, 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 July 28, 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 TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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 a margin of safety.

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

*Attorney for licensee:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

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

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

*Date of amendment request:* July 20, 2004.

*Description of amendment request:* The proposed amendments would make various changes to the technical specifications (TSs) associated with the Plant Hatch DC electrical system consistent with Technical Specification Task Force (TSTF) change traveler TSTF–360, including specific action and increased completion time for an inoperable battery charger, increase the completion time for an inoperable station service battery from 2 to 12 hours, relocate preventive maintenance surveillance requirements (SRs) to licensee controlled programs, provide alternate testing criteria for battery charger testing, replace battery specific gravity monitoring with float current monitoring, relocate and create a Section 5.5 program to reference actions for cell voltage and electrolyte level, and provide specific actions and increased completion times for out-of-limits conditions for certain battery parameters.

*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 is a proposed change to the DC system Technical Specifications. No physical changes are being proposed to any system designed to prevent a previously evaluated accident, such as a loss of coolant accident with a loss of offsite power.

This proposed TS change provides specific completion times for certain inoperable DC components, and relocates some surveillance requirements to owner controlled programs. Additionally, monitoring of specific gravity will be replaced with float current monitoring, and some Action levels for cell voltage and electrolyte level are relocated to owner controlled programs.

The completion time for battery charger inoperability is increased to 7 days; however, only after verification that the associated battery is fully operable, without such verification, the 7 day completion time is not used. Thus, adequate DC to support design basis events is ensured.

Increasing the station service battery out of service time from 2 to 12 hours will allow more time for proper maintenance to repair a faulty battery. However, the 12 hour out of service time is still a very restrictive time and so the probability of an event where the battery would be needed within this 12 hour time frame is very low. In fact, a probability risk assessment of the increased out of service time has been performed and it fell within the criteria of Reg Guide 1.174 and 1.177.

The relocation of certain SRs and action levels is done for surveillances and parameter action levels that are more intended to monitor and maintain long term component performance. These relocated items are not meant as clear levels at which the DC components can no longer be considered operable. Those that are remain in the TS. Additionally, this particular owner controlled program will be referenced in proposed Section 5.5.13 of the TS. This commitment to the program will insure that the DC system will continue to be adequately monitored and maintained.

Therefore, this proposed change to the TS ensures that the DC system will be able to provide its safety function. The probability and consequences of a previously evaluated accident are thus not increased.

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

None of the DC components will be physically altered. Furthermore, the design bases for the DC distribution systems, batteries and chargers is not changing. Although some surveillance requirements are being relocated and one (specific gravity monitoring) is being eliminated, DC system components will still be adequately surveilled and maintained. Therefore, no, new modes of operation or failure are introduced by the proposed TS change and therefore, the possibility of a new type event is not created.

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

The design functions of the DC system are unchanged. The proposed TS changes relocate many surveillance requirements and action levels to owner controlled programs. However, the owner controlled program is referenced in the new proposed Section 5.5.13 of the TS. The SNC commitment to this program will continue to ensure that the DC system is adequately monitored, surveilled, and maintained to insure that it can perform its safety function when called upon.

The addition of a 7 day completion time for the battery chargers can be used only if adequate battery capacity is verified. Thus, the DC system is capable of performing its safety function throughout the 7 day completion time.

Increasing the allowed out of service time for the station service batteries does not result in a significant reduction in the margin of safety since the proposed 12 hour time limit is still a very short time. Probabilistic risk analysis shows that the core damage frequency and large early release fractions are within the guidelines of Reg Guides 1.174 and 1.177.

Elimination of specific gravity surveillance is acceptable since the float current monitoring adequately replaces it.

For the above reasons, 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:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

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

*Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama*

*Date of amendment request:* July 2, 2004 (TS–449).

*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 July 2, 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:* General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

*NRC Section Chief:* Michael L. Marshall (Acting).

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

*Date of amendment request:* August 12, 2004.

*Description of amendment request:* The proposed Technical Specification change will revise Surveillance Requirement 4.7.8.d.3 by removing the vacuum relief flow portion. The proposed revision removes criteria from the surveillance that is not necessary to verify the operability of the Auxiliary Building Gas Treatment System (ABGTS). The bases associated with the ABGTS will be revised to remove discussions regarding the vacuum relief flow portion of this surveillance as part of this effort.

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

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

No. The proposed change removes an overly restrictive criterion for vacuum relief flow as part of the ABGTS operability verification. This criterion is not required for the verification of ABGTS operability and therefore, the removal does not reduce the associated safety function. No system modification or operating practices are changed by the proposed revision. The accident mitigation functions of the ABGTS will not be adversely affected by the proposed removal and offsite dose potential is not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed change does not result in the alteration of plant equipment or components or the modification of operating requirements for plant systems. Additionally, the ABGTS functions serve to mitigate accident conditions and are not considered a source for accident generation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed removal of an unnecessary criterion from the ABGTS surveillance will not result in a change to plant setpoints that function to maintain the safety margins. The ABGTS will continue to provide the required negative pressure conditions for the auxiliary building during accident conditions to maintain acceptable dose conditions. The actuation of safety features for accident mitigation will not be affected by the proposed changes. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

*NRC Acting Section Chief:* Michael 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 rename the Trip Setpoint column of Technical Specification (TS) Tables 2.2–1 and 3.3–4, remove the inequality signs for the trip setpoint values as appropriate, and revise the inequality representation for the allowable values, as needed. This proposed amendment is a revision to a previous amendment request dated November 15, 2002 (ADAMS Accession No. ML023290477), that supersedes the original request in its entirety.

*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:

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

The proposed revisions for the nominal trip setpoint representation are administrative changes that will not impact the application of the reactor trip or ESF [engineered safety feature] actuation system instrumentation requirements. This is based on the setpoint requirements being applied without change, as well as the Avs [allowable values], in accordance with the current setpoint methodology. The removal of the inequalities associated with the trip setpoint values will be more appropriate for the use of nominal setpoint values but will not differ in application from the setpoint methodology utilized by TVA. Deletion of the nominal terminology associated with overtemperature delta temperature average temperature at rated thermal power (T′) provides a better representation of the limit associated with this value. In addition, this change will not alter plant equipment or operating practices. Therefore, the implementation of these changes will not increase the probability or consequences of an accident.

The revision of the reactor coolant pump (RCP) underfrequency, intermediate range neutron flux P–6, and fuel storage pool area radiation monitor trip setpoints and the Avs for the RCP underfrequency, intermediate range neutron flux P–6, and undervoltage has been evaluated and the results are documented in approved calculations. These calculations verify that the revised values are acceptable in accordance with appropriate calculation methodologies and that they will continue to support the accident analysis. These revisions will not require changes to the instrumentation settings currently being used or the methods for maintaining them. The offsite dose potential will be reduced because the proposed TS values are more conservative and will ensure the adequacy of designed safety functions to limit the release of radioactivity. Therefore, the proposed revision of these values will not significantly increase the probability or consequences of an accident.

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

The revision of the nominal trip setpoint representation and elimination of the nominal nomenclature, as well as the revised setpoint values and Avs will not alter the plant configuration or functions. The revised setpoints and the proposed operability limits will continue to provide acceptable initiation of safety functions for the mitigation of postulated accidents as required by the design basis. The primary function of the reactor protection system, the ESF actuation system, and the radiation monitoring function is to initiate accident mitigation functions. These functions are not considered to be initiators of postulated accidents. The proposed changes do not create the possibility of a new or different kind of accident because the design functions are not altered and the proposed values meet the accident analysis requirements for accident mitigation.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The setpoint and Av revisions proposed in this request were evaluated and found to be acceptable without impact to the safety limits required for the associated functions. The nominal trip setpoint representation change and the elimination of inappropriate nominal indications do not alter the TS functions or their application and will not require changes to design settings. Plant systems will continue to be actuated for those plant conditions that require the initiation of accident mitigation functions. The margin of safety is not reduced because the proposed conservative changes to the Av and setpoint representations will not change design functions and the initiation of accident mitigation functions for appropriate plant conditions is ensured.

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. (Acting).

**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 email to *pdr@nrc.gov*.

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

*Date of application for amendments:* August 22, 2003.

*Brief description of amendments:* The amendments authorize revision of the Updated Final Safety Analysis Report to incorporate the description of the approved change to the maximum fuel pin pressurization criteria used in the evaluation of the design basis fuel-handling accident as described in the amendment application of August 22, 2003.

*Date of issuance:* September 27, 2004.

*Effective date:* September 27, 2004, and shall be implemented within 60 days of the date of issuance.

*Amendment Nos.:* Unit 1–153, Unit 2–153, Unit 3–153.

*Facility Operating License Nos. NPF–41, NPF–51, and NPF–74:* The amendments authorize the revision of the Updated Final Safety Analysis Report.

*Date of initial notice in* ***Federal Register:*** December 9, 2003 (68 FR 68656). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 27, 2004.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* May 10, 2002, as supplemented March 12, 2003, April 10, 2003, March 5, 2004, and July 22, 2004.

*Brief description of amendment:* The amendment approves full implementation of the alternative source term, with the exception of the loss-of-coolant accident.

*Date of issuance:* September 24, 2004.

*Effective date:* September 24, 2004.

*Amendment No.:* 201.

*Renewed Facility Operating License No. DPR–23:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** April 1, 2003 (68 FR 15758). The April 10, 2003, March 5, 2004, and July 22, 2004, supplements contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* September 26, 2002, as supplemented June 2, 2003, May 7, June 18, and August 24, 2004.

*Brief description of amendment:* The amendment revised the technical specifications (TSs) to allow relaxation of containment operability requirements while handling irradiated fuel and core alterations.

*Date of issuance:* September 20, 2004.

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

*Amendment No.:* 284.

*Facility Operating License No. DPR–65:* The amendment revised the TSs.

*Date of initial notice in* ***Federal Register:*** November 12, 2002 (67 FR 68731). The supplements dated June 2, 2003, May 7, June 18, and August 24, 2004 contained clarifying information and did not change the staff's proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* May 9, 2002, as supplemented by letter dated June 11, 2003, and August 18 and September 22, 2004.

*Brief description of amendments:* The amendments approve changes to the Updated Final Safety Analysis Report for Catawba, Units 1 and 2 to eliminate the single failure of either of the 125 VDC Distribution Centers, EDE or EDF, from the design-basis steam generator tube rupture accident analyses.

*Date of issuance:* September 24, 2004.

*Effective date:* As of the date of issuance and shall be implemented with the next update of the Safety Analysis Report in accordance with 10 CFR 50.71(e)

*Amendment Nos.:* 217, 211.

*Renewed Facility Operating License Nos. NPF–35 and NPF–52:* Amendments revise the Licensing Basis.

*Date of initial notice in* ***Federal Register:*** July 23, 2002 (67 FR 48215). The supplements dated June 11, 2003, and August 18 and September 22, 2004, provided clarifying information that did not change the scope of the May 9, 2002, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* June 30, 2003, as supplemented by letters dated November 20, 2003, February 27, 2004, and September 10, 2004.

*Brief description of amendment:* The amendment (1) reorganizes the Arkansas Nuclear One, Unit No. 2 (ANO–2) Technical Specification (TS) Section 6.0, Administrative Controls, (2) modifies the ANO–2 Facility Operating License, and Actions and Surveillance Requirements (SRs) of various other TSs, to support the reorganization of Section 6.0, and (3) modifies several Actions and SRs that are related to systems that are shared by ANO–2 and Arkansas Nuclear One, Unit No. 1.

*Date of issuance:* September 28, 2004.

*Effective date:* As of the date of issuance to be implemented within 120 days from the date of issuance.

*Amendment No.:* 255.

*Facility Operating License No. NPF–6:* Amendment revises the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** December 9, 2003 (68 FR 68663). The supplements dated February 27, 2004, and September 10, 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 September 28, 2004.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* November 25, 2002, as supplemented by letters dated May 14, 2003, September 29, 2003, and March 25, 2004.

*Brief description of amendment:* The proposed amendment would revise Technical Specification 3.9.11, ''Storage Pool Water Level'' and TS 5.6.1, ''Fuel Storage—Criticality.'' This amendment permits St. Lucie Unit 1 to credit soluble boron, fuel loading restrictions, and control element assemblies in the spent fuel pool criticality analyses and eliminate the need to credit Boraflex neutron absorbing material for reactivity control.

*Date of Issuance:* September 23, 2004.

*Effective Date:* As of the date of issuance and shall be implemented by September 30, 2005.

*Amendment No.:* 193.

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

*Date of initial notice in **Federal Register:*** January 7, 2003 (68 FR 806). The May 14, 2003, September 29, 2003, and March 25, 2004, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 23, 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:* August 25, 2003, as supplemented by letters dated February 9, February 23, March 25, April 15, May 20, and July 29, 2004.

*Description of amendment request:* The amendment revised the Technical Specifications (TSs) to extend the emergency diesel generator allowed outage time from 72 hours to a period of 14 days, and to allow extension of the current two-hour time requirement to four hours for verification of redundant component operability. These changes are in support of installing a non-safety-related supplemental emergency power system. The Bases of the affected TSs will be modified to address the changes.

*Date of issuance:* September 21, 2004.

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

*Amendment No.:* 97.

*Facility Operating License No. NPF–86:* The amendment revised the TSS.

*Date of initial notice in **Federal Register:*** December 29, 2003 (68 FR 68669). The supplements dated February 9, February 23, March 25, April 15, May 20, and July 29, 2004, did not change the staff's proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

*Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan*

*Date of application for amendments:* February 14, 2004, as supplemented July 26, 2004.

*Brief description of amendments:* The amendments modify technical specification (TS) 3.9.2 limiting condition for operation, delete TS surveillance requirements (SRs) 4.9.2.a and 4.9.2.b for the Source Range Neutron Flux Monitor channel functional test, revise SR 4.9.2.c for the channel check test, and add a requirement to perform a channel calibration every 18 months as well as revise TS 4.10.4.2 and 4.10.3.2 (Units 1 and 2 respectively) for Intermediate and Power Range channel functional test.

*Date of issuance:* September 23, 2004.

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

*Amendment Nos.:* 283, 267.

*Facility Operating License Nos. DPR–58 and DPR–74:* Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 11, 2004 (69 FR 26191).

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

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* May 25, 2004, as supplemented August 6, 2004.

*Brief description of amendment:* The amendment revises technical specification (TS) 3.10.f.2 to add an allowed outage time for the individual rod position indication (IRPI) system of 24 hours with more than one IRPI group inoperable and adds the definition of ''immediately'' to TS Section 1.0.

*Date of issuance:* September 22, 2004.

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

*Amendment No.:* 176.

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

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

The supplement dated August 6, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 2004

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* July 7, 2003, as supplemented March 17, May 18, and August 18, 2004.

*Brief description of amendment:* The amendment adds Technical Specification Section 3.3.e.1.A.3, which provides requirements for turbine building service water header isolation logic.

*Date of issuance:* September 24, 2004.

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

*Amendment No.:* 177.

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

*Date of initial notice in* **Federal Register:** August 5, 2003 (68 FR 46244).

The supplements dated March 17, May 18, and August 18, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 2004.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* July 24, 2003.

*Brief description of amendments:* The amendments revise Technical Specification (TS) 3.8.4, ''DC Sources—Operating,'' TS 3.8.5, ''DC Sources—Shutdown,'' and TS 3.8.6, ''Battery Cell Parameters,'' and add a new TS 5.5.17, ''Battery Monitoring and Maintenance Program.'' The changes adopt in part the NRC-approved Technical Specification Task Force (TSTF–360, Revision 1, ''DC Electrical Rewrite.''

*Date of issuance:* September 20, 2004.

*Effective date:* September 20, 2004, and shall be implemented within 120 days from the date of issuance. The licensee shall reflect the relocation of TS requirements to licensee-controlled programs and the TS Bases, as described in the licensee's letter dated July 24, 2003, and the NRC safety evaluation attached to the amendment, in the next scheduled update of the Final Safety Analysis Report Update submitted pursuant to 10 CFR 50.71(e).

*Amendment Nos.:* Unit 1—172; Unit 2—174.

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

*Date of initial notice in **Federal Register:*** September 2, 2003 (68 FR 52236).

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

No significant hazards consideration comments received: No.

*PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2 (SSES–2), Luzerne County, Pennsylvania*

*Date of application for amendments:* September 16, 2003, as supplemented by letter dated April 27, 2004.

*Brief description of amendments:* The amendment revised the values of the Safety Limit for Minimum Critical Power Ratio in TS 2.1.1.2 for current SSES–2 Cycle 12 mid-cycle two-recirculation-loop and single-recirculation-loop operation.

*Date of issuance:* September 21, 2004.

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

*Amendment No.:* 191.

*Facility Operating License No. NPF–22:* The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 28, 2003 (68 FR 61480). The supplement dated April 27, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 21, 2004.

No significant hazards consideration comments received: No.

*PSEG Nuclear, LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey*

*Date of application for amendments:* July 29, 2002, as supplemented by letters dated March 28, 2003, May 1, 2003, and August 20, 2004.

*Brief description of amendments:* The amendments modify the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TSs) requirements for containment closure associated with the equipment hatch and personnel airlocks during Core Alterations and movement of irradiated fuel within the containment. The change allows the equipment hatch and the personnel airlocks to remain open during fuel movement inside containment provided administrative controls are in place to ensure the closure of the equipment hatch and personnel airlock following a fuel handling accident within the containment building. In addition, the associated TS Bases are revised.

*Date of issuance:* September 16, 2004.

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

*Amendment Nos.:* 263 and 245.

*Facility Operating License Nos. DPR–70 and DPR–75:* The amendments revised the TSs.

*Date of initial notice in **Federal Register:*** August 20, 2002 (67 FR 53989). The licensee's supplements dated March 28, 2003, May 1, 2003, and August 20, 2004, provided clarifying information that did not change the scope of the proposed amendments as described in the original notice of proposed action published in the **Federal Register**, and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 16, 2004.

No significant hazards consideration comments received: No.

*Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York*

*Date of application for amendment:* April 9, 2002, as supplemented January 10, 2003, February 24, 2004, and August 27, 2004.

*Brief description of amendment:* The amendment revised the Ginna Improved Technical Specification with regards to: relocating figures associated with Core Safety Limits to the Core Operating Limits Report (COLR), relocating Overtemperture $\Delta T$ and Overpower $\Delta T$ parameters to the COLR, and replacing current trip setpoints for the Reactor Protection System and the Engineered Safety Feature Actuation System with Limiting Safety System Settings in accordance with the Instrument Society of America Standard 67.04, Part 2.

*Date of issuance:* September 22, 2004.

*Effective date:* As of the date of issuance to be implemented within 1 year.

*Amendment No.:* 85.

*Renewed Facility Operating License No. DPR–18:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 28, 2002 (67 FR 36933). The supplements dated January 10, 2003, February 24, 2004 and August 27, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* January 28, 2004, supplemented by a letter dated July 23, 2004.

*Brief description of amendment:* The amendment revises the SONGS Unit 1 License and Permanently Defueled Technical Specifications to modify or remove operational and administrative requirements that are not applicable upon the transfer of all spent fuel from the spent fuel pool into the SONGS dry cask storage Independent Spent Fuel Storage Installation.

*Date of issuance:* September 21, 2004.

*Effective date:* As of the date that all reactor fuel has been permanently removed from the spent fuel pool and stored in an Independent Spent Fuel Storage Installation. The license amendment shall be implemented within 30 days of its effective date.

*Amendment No.:* 163.

*Facility Operating License No. DPR–13:* This amendment revises both the license and the technical specifications.

*Date of initial notice in* ***Federal Register:*** March 30, 2004 (69 FR 16623). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 21, 2004.

No significant hazards consideration comments received: No.

*Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama*

*Date of application for amendments:* July 31, 2002 as supplemented by letters dated December 9, 2002, February 12, 2003, March 26, 2003, July 11, 2003, July 17, 2003, May 17, 2004, July 2, 2004, August 24, 2004 and September 17, 2004.

*Description of amendment request:* The amendments requested full implementation of an alternative source term (AST) methodology for the Units 1, 2, and 3 operating licenses and design bases. The amendments adopt the AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67. The submittals also proposed to revise and/or remove the Technical Specification (TS) Sections associated with control room emergency ventilation (CREV), standby gas treatment (SGT), standby liquid control (SLC), and secondary containment systems. Additionally, the submittals requested modification of the licensing and design basis to reflect the application of the AST methodology and the function of the SLC system, and deletion of a license condition for Units 2 and 3.

The supplements to the original application included the withdrawal of the request to delete one of the TS Sections described above, associated with the absorption of elemental iodine by the SGT and CREV systems charcoal filters. Also the supplements added a new TS Section to require verification that the minimum fuel decay period has passed prior to moving fuel after the reactor is shut down.

*Date of issuance:* September 27, 2004.

*Effective date:* Date of issuance, to be implemented prior to restart of Unit 1, and within 120 days for Units 2 and 3.

*Amendment Nos.:* 251, 290 and 249.

Facility Operating License Nos. *DPR–33, DPR–52, and DPR–68:* Amendments revised the Operating Licenses and TSs.

*Date of initial notice in* ***Federal Register:*** October 15, 2002 (67 FR 63697). The supplements dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, provided information that changed the scope of the original request, therefore another **Federal Register** notice was published on April 27, 2004 (69 FR 22883). However, the supplements dated May 17, July 2, August 24, and September 17, 2004, provided clarifying information that did not expand the scope of the revised request or the proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* March 5, 2004.

*Brief description of amendment:* The amendments delete surveillance requirements to perform certain channel functional tests of the source range, intermediate, and power range neutron flux monitors. These amendments eliminate extraneous and unnecessary performance of these surveillances.

*Date of issuance:* September 20, 2004.

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

*Amendment Nos.:* 295 and 285.

*Facility Operating License No. DPR–77 and DPR–79:* Amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register:** April 13, 2004 (69 FR 19576). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 2004.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* March 5, 2004.

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

*Date of issuance:* September 20, 2004.

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

*Amendment Nos.:* 296 and 286.

*Facility Operating License Nos. DPR–77 and DPR–79:* Amendments revised the technical specifications.

*Date of initial notice in* ***Federal Register:*** April 13, 2004 (69 FR 19576).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 1st day of October, 2004.

For the Nuclear Regulatory Commission.

**William H. Ruland,**

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

[FR Doc. 04–22544 Filed 10–8–04; 8:45 am]

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## SECURITIES AND EXCHANGE COMMISSION

**[Release No. 34–50478; File No. SR–NASD–2004–107]**

## Self-Regulatory Organizations; National Association of Securities Dealers, Inc.; Notice of Filing of a Proposed Rule Change Relating to Computer Generated Quoting in Exchange-Listed Securities

September 30, 2004.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] notice is hereby given that on July 12, 2004, the National Association of Securities Dealers, Inc. (''NASD''), through its subsidiary, The Nasdaq Stock Market, Inc. (''Nasdaq''), filed

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.