Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Because of continuing disruptions in the delivery of mail to United States Government offices, it is requested that requests for hearing also be transmitted to the Office of the General Counsel, either by means of facsimile transmission to 301–415–3725, or by e-mail to *OGCMailCenter@nrc.gov.*

In addition to meeting other applicable requirements of 10 CFR Part 2 of the NRC's regulations, a request for a hearing filed by a person other than an applicant must describe in detail:

(1) The interest of the requestor;

(2) How that interest may be affected by the results of the proceeding, including the reasons why the requestor should be permitted a hearing, with particular reference to the factors set out in § 2.1205(h);

(3) The requestor's areas of concern about the licensing activity that is the subject matter of the proceeding; and

(4) The circumstances establishing that the request for a hearing is timely in accordance with § 2.1205(d).

III. Further Information

The application for the license amendment and the request to revise the License Application are available for inspection at NRC's Public Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* Documents may also be examined and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike, Rockville, MD 20854. Any questions with respect to this action should be referred to Myron Fliegel, Fuel Cycle Facilities Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Mail Stop T8–A33, Washington, DC 20555–0001. Telephone: (301) 415–6629.

Dated at Rockville, Maryland, this 8th day of April, 2003.

For the Nuclear Regulatory Commission.

Lidia Roché,

Acting Chief, Fuel Cycle Facilities Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 03–9197 Filed 4–14–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATE: Weeks of April 14, 21, 28, May 5, 12, 19, 2003.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of April 14, 2003

There are no meetings scheduled for the Week of April 14, 2003.

Week of April 21, 2003—Tentative

There are no meetings scheduled for the Week of April 21, 2003.

Week of April 28, 2003—Tentative

There are no meetings scheduled for the Week of April 28, 2003.

Week of May 5, 2003—Tentative

There are no meetings scheduled for the Week of May 5, 2003.

Week of May 12, 2003—Tentative

Thursday, May 15, 2003

9:30 a.m.—Briefing on results of Agency Action Review Meeting (Public Meeting) (Contact: Robert Pascarelli, 301–415–1245)

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

Week of May 19, 2003—Tentative

There are no meetings scheduled for the Week of May 19, 2003.

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

* * * * *

Additional Information

By a vote of 4–0 on April 8, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Security Issues (Closed—Ex. 1)'' be held on April 9, and on less than one week's notice to the public.

By a vote of 4–0 on April 8 & 9, the Commission determined pursuant to U.S. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Security Issues (Closed—Ex. 1)'' be held on April 11, and on less than one week's notice to the public.

* * * * *

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

* * * * *

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

Dated: April 10, 2003.

D.L. Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–9311 Filed 4–11–03; 11:24 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 4, 2003, through April 17, 2003. The last biweekly notice was published on April 1, 2003, (68 FR 15756).

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.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing 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 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By May 15, 2003, 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 2.714, 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 by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, 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 factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide 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 to relief. A petitioner who fails to file such a supplement which satisfies 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, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, 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 with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* 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 because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1 (Fermi 1), Monroe County, Michigan

Date of amendment request: January 28, 2003, (Reference NRC–03–0011).

Description of amendment request: The proposed amendment will revise the Technical Specifications by:

1. Section A.1, 2, 4, 8, C.1, D, E.1, H.3.b, I.5, I.7b, I.9.d have been previously deleted and the word ''Deleted'' used as a place marker to alleviate the need to renumber all sections. This request proposes to remove these sections and renumber as appropriate.

2. Sections C.2 and E.2 cover the Reactor Building and Fuel and Repair Building Drains. This request proposes to delete the requirements in sections C.2 and E.2, which is all that remains in sections C and E. Section C, Reactor Building, and E, Fuel and Repair Building, will be deleted in their entirety.

3. Added, ''Monitoring or sampling for tritium will not be required if the sample results have determined that tritium is not present during a given evolution'' in Section F. This is to clarify the intent of ''During other evolutions resulting in radioactive gaseous effluents, the effluents shall be monitored or sampled and analyzed for tritium and particulates.''

4. Section H.1 and 2 cover alarms, including surveillances, allowed out of service time, compensatory measures and alarm readouts for alarms associated with water intrusion. This request proposed to delete these sections on water intrusion alarms.

5. Sections H.3 and 4 cover required inspections of the facility. This request proposes to delete the requirement for radiation surveillance of the steam cleaning room access plug, which is Item c. of H.3, Fuel and Repair Building.

This proposal adds the words ''(until made inactive)'' to H.3 Reactor Building Item c. This request also proposes to delete recording liquid waste tank levels, which is Item c. in Section H.4.

6. Table H–1 lists the required Fermi 1 alarms and their alarm points. Only water intrusion alarms are currently covered in this table. This request proposed to delete this alarm table.

7. Editorial changes are included in this proposed request. In section I.2, the word ''employes'' will be changed to ''employees''. In Section I.2.b the word ''He'' will be changed to ''The Health Physicist''. In Section I.7 the word ''his'' will be removed from the following sentence, ''The Custodian may temporarily change a procedure by Written Order following his determination that the change does not constitute a significant increase in the hazards associated with the operation.'' In Section I.9.h the word ''usual'' will be changed to ''unusual''.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

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

Removing the requirements for water intrusion monitoring, liquid waste tanks level recording, and building drains will not significantly increase the possibility of an accident as long as the probability of an uncontrolled sodium and water reaction is not significantly increased. This is accomplished by the amount of volume of the area in which the sodium is present where water intrusion is currently monitored. It would take a long period of time for the water intrusion to reach the sodium piping and this would still not increase the probability as long as the piping is not breached. When the piping is breached during the sodium abatement process, it will be completed under controlled conditions. Removal of the instrumentation may delay the discovery of a liquid spill but cannot affect the probability of the spill since it is only instrumentation. The consequences of an accident will not be increased because the previously analyzed accident accounts for all of the radioactive material contained within the liquid waste tanks and primary sodium to be released. This change will not increase the amount of radioactive material. The editorial changes, steam cleaning room plug radiation survey deletion, or the clarification made to gaseous effluent monitoring for tritium will not significantly increase the probability or consequences of an accident, because they have no impact on how any systems are operated or what systems are removed from the facility.

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

Removing the requirements for water intrusion monitoring and liquid waste tanks level recording will not create the possibility of a new or different accident from any previously evaluated. The accidents these systems monitor for have already been analyzed for, including a release of the radioactive sodium during a sodium and water reaction and the release of the entire contents of the liquid waste tanks. Removing the building drains requirements will not cause a different type of accident since the drains only affect where liquid flows. Where liquid flows cannot cause an accident unless the drains place water where it does not belong. This can only impact a liquid water release or sodium accident. The editorial changes, survey deletion, and the clarification made to gaseous effluent monitoring for tritium will not create the possibility of a new or different accident, since they do not introduce any new modes of operation of facility equipment.

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

The removal of the requirements for water intrusion monitoring, liquid waste tanks level recording, and building drains may slightly reduce the margin of safety, but not significantly. Removing them does not in itself introduce water into the sodium containing systems. Nor does removing them allow for an unmonitored discharge of any radioactive effluents. Discharges are still controlled by Section C of the proposed amendment to the Technical Specifications. The decommissioning project is now ongoing and the facility no longer normally vacant as it was during the initial time following facility retirement. In addition, the calculated consequences of releasing the radioactive material are small and within 10 CFR 20 limits. The editorial changes or survey deletion will not significantly reduce a margin of safety, because the survey is of a floor plug that has been removed from the entrance to an area and has no function. The clarification made to gaseous effluent monitoring for tritium will not significantly reduce a margin of safety since tritium monitoring is still required for evolutions involving sodium processing and pipe cutting, and during other activities, unless results have determined tritium is not present during a given evolution.

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

Attorney for licensee: John Flynn, Esquire, Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig.

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 amendment request: November 25, 2002.

Description of amendment request: The amendments would revise the Technical Specifications (TS) for the Ventilation Filter Testing Program (VFTP), Annulus Ventilation System (AVS), Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Fuel Handling Ventilation Exhaust System (FHVES), and Control Room Area Ventilation System (CRAVS), and containment penetrations.

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:

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

This licensee amendment request proposes amendments to the system TS and/or Bases and/or VFTP TS requirements for the AVS, ABFVES, FHVES, and CRAVS. It also proposes amendments to the TS and Bases for Containment Penetrations. The AVS is in standby during normal plant operations and operates only following a Safety Injection signal or during a test. It is not an accident initiator. The ABFVES is in operation during normal plant operations. However, the ABFVES is not used in direct support of any phase of power generation or conversion or transmission, shutdown cooling, fuel handling operations, or processing of radioactive fluids. Therefore, it is not an accident initiator. The FHVES is utilized to support fuel handling operations when moving recently irradiated fuel. It is not an accident initiator. The CRAVS operates during normal plant operations. However, it is not an accident initiator (the CRAVS being defined so as to exclude equipment that maintains an appropriately low temperature in the control room). The status of containment penetrations is required to be controlled so as to minimize the consequences of a fuel handling accident or a weir gate drop accident. The containment penetrations by themselves are not accident initiators. No accident initiators are associated with the changes proposed in this license amendment request. For these reasons, operation of the facility in accordance with this proposed amendment does not involve a significant increase in the probability of any accident previously evaluated.

In support of the proposed amendment, an analysis has been performed to determine the radiological consequences of the design basis LOCA [loss-of-coolant accident] at Catawba Nuclear Station. The analysis made use of the Alternative Source Term (AST) methodology and in general conformed to the regulatory positions of Regulatory Guide 1.183, [''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' (ML003716792)

(Draft DG1081 Issued December 1999)] and the draft regulatory positions of DG–1111. Total Effective Dose Equivalent (TEDE) radiation doses at the Exclusion Area Boundary (EAB), boundary of the Low Population Zone (LPZ), and to the control room operators were calculated and found to be acceptable.

TEDE's have been estimated from the radiation doses with the current analysis (reported in the UFSAR [Updated Final Safety Analysis Report]) using the guidelines of Regulatory Guide 1.183 modified as reported in Appendix A of Attachment 3 [of the licensee's submittal dated November 25, 2002]. These TEDE's are compared to the limiting TEDE's from the proposed analysis as follows:

TEDE'S FOLLOWING THE DESIGN BASIS LOCA

Location	TEDE'S (Rem)	
	UFSAR	Proposed
EAB	9.95	7.21
LPZ	1.90	3.97
Control Room	1.57	2.65

The new value for the control room TEDE radiation dose is higher than the TEDE radiation dose equivalent to the radiation doses currently reported in the UFSAR. However, the limiting control room TEDE radiation dose reported in this submittal is lower than the acceptance criterion by 47%. The new LPZ TEDE radiation dose is higher than the equivalent TEDE radiation dose currently represented. On the other hand, the margin to the acceptance criterion is 84%. The TEDE radiation doses newly computed at the EAB for the design basis LOCA is lower than the corresponding equivalent EAB TEDE radiation dose currently represented in the UFSAR. The margin in the EAB TEDE radiation dose to the guideline value is 71%. In all cases, there is significant margin between the newly calculated post-LOCA TEDE radiation doses and the corresponding regulatory guideline values. In the sense that the margins to the germane regulatory guideline values are still large, the new values of TEDE radiation doses are comparable to the equivalent TEDE associated with the post-LOCA radiation doses currently listed in the UFSAR. Therefore, the proposed amendment is determined to not result in a significant increase in accident consequences.

The changes proposed to the TS for Containment Penetrations are editorial in nature and will have no effect upon accident consequences.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant increase in any accident consequences. The changes to make the penetration values for Unit 2 consistent with Unit 1 for the AVS, ABFVES, and FHVES are acceptable because the appropriate safety factors as delineated in the applicable regulatory guideline documents are still maintained. The change to the flowrate specified for the ABFVES is consistent with the design basis operation of this system. Also, the editorial changes proposed to the VFTP TS will have no impact on any accidents.

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

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

This proposed amendment does not involve addition, removal, or modification of any plant system, structure, or component. These changes will not affect the operation of any plant system, structure, or components as directed in plant procedures.

The analysis performed in support of this license amendment request, together with the analyses of the design basis fuel handling accident and weir gate drop reported in previously submitted and NRC approved license amendment requests, includes full scope implementation of AST methodology. This analysis does not represent any change in the post-accident operation of any plant system, structure, or component.

Operation of the facility in accordance with this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Margin of safety is related to confidence in the ability of fission product barriers to perform their design functions following any of their design basis accidents. These barriers include the fuel cladding, the Reactor Coolant System, and the containment. The performance of these barriers either during normal plant operations or following an accident will not be affected by the changes associated with the license amendment request.

The AVS is associated with the containment fission product barrier. Its post-accident operation will not be affected by implementation of the amendment to its TS. The operation of the ABFVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the FHVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the CRAVS either during normal plant operations or following an accident will not be adversely affected by the proposed changes to its TS Bases. The operation of Containment Penetrations following an accident will not be adversely affected by the proposed change to its TS.

As noted, an analysis of radiological consequences of the design LOCA at Catawba Nuclear Station has been performed in support of this license amendment request. The design basis LOCA scenarios were selected based on extensive evaluations of Catawba, its design basis, and its anticipated response to a design basis LOCA. Credit was taken only for safety related systems, structures, and components in simulating the mitigation of radiological consequences of the LOCA. Limiting values were taken for performance characteristics of the Class 1E systems modeled in the analysis. The radiological consequences (TEDE radiation doses at the EAB, LPZ, and in the control room) are within the regulatory guideline values with significant margin.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant reduction in the margin of safety. These changes are supported by regulatory guidance documents, and are consistent with existing system operation. Also, the editorial changes proposed to the VFTP TS will not have any impact on safety.

Operation of the facility in accordance with the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: November 20, 2002, as supplemented January 21, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for REQUIRED ACTIONS requiring suspension of operations involving positive reactivity additions and various NOTES that preclude reduction in boron concentration. The proposed changes revise these REQUIRED ACTIONS and NOTES to limit the introduction of positive reactivity such that the required margin to criticality, the shutdown margin and refueling boron concentration limits will still be satisfied. The licensee stated that the changes are consistent with the Technical Specification Task Force (TSTF) traveler number 286, Revision 2. Associated changes are also proposed for the TS Bases. *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:

The following discussion is a summary of the evaluation of the change contained in this proposed amendment against the 10 CFR 50.92 (c) requirements to demonstrate that all three standards are satisfied. A ''no significant hazards consideration'' is indicated if operation of the facility in accordance with the proposed amendment would not:

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

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

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

First Standard

The proposed changes do not involve any physical alteration of plant systems, structures, or components. The proposed changes revise ACTIONS in the Catawba Nuclear Station (CNS) and McGuire Nuclear Station (MNS) Technical Specifications (TS) that require suspending operations involving positive reactivity additions and several Limiting Condition for Operation (LCO) Notes that preclude reduction in boron concentration. The change revises these ACTIONS and LCO Notes to limit the introduction of reactivity such that the required SHUTDOWN MARGIN (SDM) or refueling boron concentration will still be satisfied. The proposed change ensures that the reactivity condition [k_{eff}] specified in mode definition, the SDM of LCO 3.1.1 and minimum boron concentration requirements of LCO 3.9.1 are met. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated in the updated final safety analysis report (UFSAR) because the accident analysis assumptions and initial conditions will continue to be maintained.

Second Standard

The proposed changes do not involve any physical alteration of plant systems, structures, or components. The proposed changes, which allow positive reactivity additions that do not result in the SDM or the refueling boron concentration being exceeded, do not introduce new failure mechanisms for system structures, or components not already considered in the UFSAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created because no new failure mechanisms or initiating events have been introduced.

Third Standard

The proposed changes do not involve a significant reduction in a margin of safety because the ability to make the reactor subcritical and maintain it subcritical during all operating conditions and modes of operation will be maintained. The margin of safety is defined by the SDM of LCO 3.1.1 and minimum boron concentration requirements of LCO 3.9.1. The proposed changes do not affect these operating restrictions and the margin of safety, which assures the ability to make and maintain the reactor subcritical, is not affected.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: January 31, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to incorporate an asymmetrical ice mass distribution within the ice condenser containment (ICC) by specifying revised safety analysis ice mass quantity requirements for three specific radial zones of the ice bed. Associated changes to the Bases were also proposed.

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:

Duke Energy Corporation (Duke) has concluded that operation of Catawba Nuclear Station (CNS) Units 1 & 2, and McGuire Nuclear Station (MNS) Units 1 & 2, in accordance with the proposed changes to the Technical Specifications (TS) does not involve a significant hazards consideration. Duke's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The Proposed Change Does Not Involve a Significant Increase In The Probability or Consequences Of An Accident Previously Evaluated.

The only analyzed accidents of possible consideration in regards to changes potentially affecting the ice condenser are a loss of coolant accident (LOCA) and a high energy line break (HELB) inside containment. However, the ice condenser is not postulated as being the initiator of any LOCA or HELB. That is because it is designed to remain functional following a design basis earthquake, and the ice condenser does not interconnect or interact with any systems that interconnect or interact with the Reactor Coolant or Main Steam Systems. Since these proposed changes do not result in, or require, any physical change to the ice condenser that could introduce an interaction with the Reactor Coolant or Main Steam Systems, then there can be no change in the probability of an accident previously evaluated.

Regarding consequences of analyzed accidents, the ice condenser is an engineered safety feature designed, in part, to limit the containment sub-compartment and containment vessel pressure immediately following the initiation of a LOCA or HELB. Conservative sub-compartment and containment pressure analysis [based on the proposed changes] shows these criteria will be met if the total ice mass within the ice bed is maintained in accordance with the DBA [Design Basis Accident] analysis; therefore, the proposed TS SR [Surveillance Requirement] changes of these requirements will not increase the consequences of any accident previously evaluated.

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

B. The Proposed Change Does Not Create The Possibility Of A New Or Different Kind Of Accident From Any Accident Previously Evaluated.

As previously described, the ice condenser is not postulated as being the initiator of any design basis accident. The proposed changes do not impact any plant system, structure or component that is an accident initiator. The proposed TSs and TS Bases changes do not involve any hardware changes to the ice condenser or other change that could create any new accident mechanisms. Therefore, there can be no new or different accidents created from those already identified and evaluated.

C. The Proposed Change Does Not Involve A Significant Reduction In A Margin Of Safety.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fuel cladding and the reactor coolant system will not be impacted by the proposed changes. The Application provides a description of additional sub-compartment and containment pressure response analysis that has been performed. This analysis demonstrates that containment will remain fully capable of performing its design function with implementation of the proposed changes. Therefore, no safety margin will be significantly impacted.

Ice Condenser plant historical operating experience has shown that the condition of the ice condenser can be ensured to be fully capable of performing its specified safety functions with performing ice mass verifications and ice mass distribution SRs on an 18 month frequency. The request to increase the MNS [McGuire] surveillance interval from 9 months to 18 months will provide performance of ice mass verification at the end of the fuel cycle, which will verify that the maintenance program is effective in maintaining the ice mass for the entire fuel cycle. Duke's utilization of the data from previous performance of TS required ice mass inspections, and additional inspection beyond these requirements, has enabled the development of a maintenance program that is reliably predictive regarding the specific operating characteristics of each [of] the ice beds at Catawba and McGuire Nuclear Stations. This maintenance program reliably predicts sublimation and determines which ice baskets to replenish prior to beginning a new 18 months operating cycle. An ice mass surveillance performed at the conclusion of the 18 month frequency in an as-found condition verifies that the maintenance program is restoring the ice bed operating cycle to maintain the ice mass quantity and distribution requirements for performance of the intended safety functions.

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

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

NRC Section Chief: John A. Nakoski.

Duke Energy Corporation, Docket No. 50–370, McGuire Nuclear Station, Unit 2, Mecklenburg County, North Carolina

Date of amendment request: January 31, 2003.

Description of amendment request: The proposed amendment would authorize the licensee to change the Updated Final Safety Analysis Report (UFSAR) to describe a process for the intentional puncture of an irradiated fuel rod in order to transfer the fuel rod gap gasses to a collection chamber, and then straighten the fuel rod for storage in a broken rod capsule.

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:

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of amendment,'' as discussed below:

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

The bent rod, located in the McGuire Unit 2 spent fuel pool, has no interfaces with any primary system, secondary system, or power transmission system. All work will be performed in the spent fuel pool, with the bent rod located under approximately 23 feet of water. None of the systems listed above are modified by the activity. No accident initiator or accident mitigation systems, for any UFSAR [Updated Final Safety Analysis Report] Chapter 15 accidents, other than fuel handling accidents, are affected with this proposed procedure for degassing and straightening of the irradiated Mk-BW fuel rod. For these reasons, the activity does not involve an increase in the probability of an accident previously evaluated.

This evolution is bounded by the UFSAR Chapter 15 dropped fuel assembly fuel handling accident inside the fuel handling building. This accident assumes that the postulated accident occurs 100 hours after reactor shutdown, the fuel assembly had 60 GWD/MTU [Gigawatt Days/Metric Ton Uranium] burnup, all rods in one fuel assembly are ruptured, and the assembly damaged has the highest peaking factor. The resultant Exclusion Area Boundary doses for the UFSAR Chapter 15 accident are 0.8 Rem Whole Body and 9.1 Rem Thyroid.

For the planned evolution, the cladding on only one rod will be breached and the fission product gas contained. This evolution will occur approximately ten years after reactor shutdown. The fuel rod burnup is only 20.46 GWD/MTU, and the fuel pin peaking factor is 1.28. Some accident mitigation will be provided by the fuel building ventilation system filters, although the majority of the activity will be from Kr-85, a noble gas, which is unaffected by these filters. The highest potential dose occurs to a worker in the fuel building, with whole body doses of less than 3 mRem and a thyroid dose of less than 3E–11 mRem. Doses at the Exclusion Area Boundary are trivial.

Should the gas container fail, the offsite activity release and, as such, the consequences of this accident will be less than any previously evaluated. Analyses have been performed to determine upper bounds for the source term, the offsite doses, and the control room dose. Both the source term and doses were found to be significantly lower than the results of the corresponding design basis analyses.

For the above reasons, it is determined that the intentional degassing of the Mk-BW fuel rod does not involve a significant increase in either the probability or the 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?

As discussed above, no ''accident initiators'' are affected by the proposed activity. The planned evolution is bounded by the dropped fuel assembly fuel handling accident inside the fuel handling building. The fuel rod straightening and degassing tools are no heavier than other fuel handling tools utilized in the spent fuel pool during routine operations. A safety tray will be placed on top of the racks and below the work area to capture any falling debris during the operation. Also a mockup operation will be performed at the Framatome facilities to identify and correct any deficiencies in the tools and processes.

For these reasons, the activity will not create the possibility of a new or different type of accident from any previously evaluated.

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

Margin of safety is associated with the confidence in the ability of the fission product barriers (the fuel and fuel cladding, the reactor coolant system pressure boundary, and the containment) to limit the level of radiation doses to the public. The proposed degassing of the fuel rod will intentionally breach the fuel rod cladding, but the fuel rod gap gasses will be captured in a collection chamber for holdup and later controlled release.

This evolution will occur beyond a nine year cooling and isotopic decay period. The level of activity in the fuel rod is very low compared to the level of activity associated with the postulated fuel handling accident; the only significant activity remaining is approximately 10 Ci [Curies] of Krypton 85. The bent rod will be maintained under 23 feet of water. Should the collection chamber fail, and the fuel rod gap gas activity released, the highest potential dose occurs to a worker in the fuel handling building, with whole body doses of less than 3 mRem, and a thyroid dose of less than 3E–11 mRem. For this reason, the resulting dose to the public is inconsequential. Both offsite doses and doses to the control room were found to be small compared to the limits of 10 CFR 100 and GDC 19. For these reasons, the activity 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: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 14, 2003.

Description of amendment request: The licensee requests modification of the River Bend Technical Specifications to revise several of the Surveillance Requirements (SRs) pertaining to testing of the Division 1 and 2 standby diesel generators (DGs). The proposed change would modify specific restrictions associated with these SRs that prohibit performing required testing in Modes 1 and 2. The affected SRs are SR 3.8.1.9 and SR 3.8.1.10.

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

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

Response: No.

The DG and its associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. As such, the ability of the DG to respond to a design basis accident will not be adversely impacted by these proposed changes. The capability of the DG to supply power in a timely manner will not be compromised by permitting performance of DG testing during periods of power operation. Additionally, limiting testing to only one DG at a time ensures that design basis requirements for backup power is met, should a fault occur on the tested DG. Therefore, there would be no significant impact on any accident consequences.

Based on the above, the proposed change to permit certain DG surveillance tests to be performed during plant operation will have no effect on accident probabilities or consequences. 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.

No new accident causal mechanisms would be created as a result of NRC [U.S. Nuclear Regulatory Commission] approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

Based on the above, implementation of the proposed changes would 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.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the DG do not affect the operability requirements for the DG, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the DG to perform its required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers.

Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 27, 2003.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TS) and other elements of the licensing bases 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, ''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 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 changes are based on Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** (FR) on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, 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 FR on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated February 27, 2003.

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 Specifications (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.

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Richard J. Laufer.

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: March 20, 2003.

Description of amendment request: This proposed change reflects an expanded operating domain for Vermont Yankee Nuclear Power Station (VY) resulting from the proposed implementation of the Average Power Range Monitor, Rod Block Monitor Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA).

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves allowing VY to operate in an expanded operating domain. Physical changes provide for enhanced instrument performance or were the result of safety analyses that support mitigation of design bases accidents. There are no changes to radioactive source terms or release pathways. The proposed change does not result in any significant change in the availability of logic systems or safety-related systems themselves. Required protective functions will be maintained. The proposed change does not degrade plant design, operation, or the performance of any safety system assumed to function in the accident analysis.

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

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

The proposed change, which allows VY to operate in an expanded operating domain, does not introduce any new accidents or failure mechanisms because the change and the effects on existing structures, systems and components have been evaluated and found to not have any adverse effects. The proposed change will not substantially impose new requirements or eliminate any existing requirements.

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

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

The proposed change, which allows VY to operate in an expanded operating domain, does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There is no impact on the conclusions of any safety analysis. The proposed change does not involve any increase in calculated off-site dose consequences. The performance of equipment will not be significantly affected.

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

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: January 31, 2003.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) allowable values (AVs) for isolation condenser system isolation Function 4.a, Steam Flow-High, and Function 4.b, Return Flow-High.

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

The proposed TS changes support the replacement of a differential pressure switch with a functionally equivalent differential pressure switch. Since there are no functional changes and no change in analytical limits, there is no significant increase in the probability or consequences of an accident previously evaluated.

Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite as a result of the proposed change.

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

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change does not adversely impact the manner in which the instrument will operate under normal and abnormal operating conditions. Therefore, these changes provide an equivalent level of safety and will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in allowed values do not affect the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The proposed changes do not affect the probability of failure or availability of the affected instrumentation. The revised AVs do not affect the analytical limits assumed in the safety analyses for actuation of instrumentation. Therefore, the proposed changes do not result in a 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: February 17, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (ITS) 3.6.3 ''Containment Isolation Valves,'' to allow verification by administrative means of isolation devices in high radiation areas, and isolation devices that are locked, sealed or otherwise secured. The specific Conditions and Surveillance Requirements (SR) in ITS 3.6.3 that will be affected are: (1) Condition A—Required Action A.2, (2) Condition B—Required Actions B.1 and B.2, (3) Condition C—Required Action C.2, and (4) SR 3.6.3.3 and SR 3.6.3.4. The licensee stated that the changes are consistent with the NUREG–1430, ''Standard Technical Specifications: Babcock and Wilcox Plants,'' Revision 2, and Standard Technical Specification Task Force (TSTF) Traveler TSTF–440. Associated changes are also proposed for the ITS Bases.

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

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

The proposed License Amendment Request (LAR) will revise the position verification requirements for manual containment isolation devices that are locked, sealed, or otherwise secured in the closed position. The proposed changes will allow the use of administrative controls to verify the position of these types of devices when they are being used to meet the Required Actions of ITS 3.6.3 Condition A, Condition B or Condition C, and will exclude these valves from Surveillance Requirement (SR) 3.6.3.3 and

SR 3.6.3.4 physical position verification requirements.

The design function of the affected containment isolation valves, and the initial conditions for accidents that require these valves to be closed, will not be affected by the proposed changes. Therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed license amendment will revise the position verification requirements for manual containment isolation devices that are locked, sealed, or otherwise secured in the closed position.

No changes to the actual position/status of these valves are proposed by this amendment. The proposed amendment will not result in changes to the design, physical configuration or operation of the plant. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changes to the position verification requirements of normally closed manual containment isolation valves that are locked, sealed, or otherwise secured do not change the position/status of these valves. The proposed amendment does not impact the ability of these valves to perform their design function of controlling containment leakage rates during design basis radiological accidents. Therefore, the proposed amendment does not result in a reduction of the margin of safety.

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602–1551.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: March 11, 2003.

Description of amendment request: The proposed amendment would change the operating license to authorize the licensee to revise the updated final safety analysis report (UFSAR) by deleting a footnote stating that the Nuclear Regulatory Commission (NRC) does not endorse the reactor building crane as single-failure-proof.

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

For heavy load handling associated with the spent fuel pool, Section 5.1.4(2) of NUREG–0612 states ''The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 are satisfied.''

An alternative to this is Section 5.1.4(1): ''The reactor building crane, and associated lifting devices used for handling of * * * heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.''

The upgraded crane and handling systems satisfy the guidelines of Section 5.1.6. The evaluation criteria of NUREG–0612, Section 5.1 are met with a single-failure-proof crane that satisfies the guidelines of Section 5.1.6, or consequence analysis that satisfies Section 5.1.4(2).

Section 5.2 of NUREG–0612 states that an evaluation of fault trees shows that: ''(1) The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after implementation of the guidelines of Section 5.1 is very low; and (2) The potential for unacceptable consequences is comparable for any of the alternatives evaluated for fault trees, indicating the relative equivalency between alternatives.''

Since the NRC fault tree evaluation shows that the potential for unacceptable consequences is comparable for the two alternatives in Section 5.1.4 of NUREG–0612, the proposed request does not significantly change the potential for unacceptable consequences to the plant in conducting heavy load handling above the spent fuel pool. The probability of a load drop accident caused by use of the reactor building crane has been reduced to where it is so small to be considered not credible within regulatory accepted standards. The reason for this is attributed to the following:

(a) The reactor building crane is single-failure-proof. In 1985, the DAEC [Duane Arnold Energy Center] Reactor Building Crane was modified to meet the requirements of NUREG–0554 ''Single Failure Proof Cranes for Nuclear Power Plants.'' The design of the Ederer hoist and trolley system was evaluated in a Staff SER [Safety Evaluation Report] of the Generic Licensing Topical Report EDR–1, Rev. 3, for Ederer's Nuclear Safety-Related Extra Safety and Monitoring (X–SAM) Cranes, dated August 3, 1983.

(b) The rigging used with the crane will be single-failure-proof per Section 5.1.6 of NUREG–0612.

(c) The requirements of NUREG–0612 Phase 1 have been implemented. The NRC provided a Safety Evaluation (SE) and Technical Evaluation Report (TER) by letter dated June 12, 1984 that concluded that the guidelines of NUREG–0612, Sections 5.1.1 and 5.3 had been satisfied and that Phase I of this issue for the DAEC was acceptable.

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

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

The crane has been upgraded to meet single-failure-proof requirements in accordance with the applicable provisions of NUREG–0612 and NUREG–0554. The use of a single-failure-proof crane with rigging and procedures that implement the requirements of NUREG–0612 assures that a cask drop is not credible. The loading on the single-failure-proof crane will not exceed the design rated load of the crane.

Rigging for critical loads will meet NUREG–0612 requirements for single-failure-proof handling systems whenever a critical load is to be lifted over safety related equipment, or over the spent fuel pool, or over the cask when it is in the reactor building and loaded with fuel. When a cask is loaded on the crane hook, the crane trolley and bridge movements will be maintained within well defined limits of operation.

The loading conditions, load combinations, allowable stress limits, and methods of analysis used in the evaluations are consistent with the current licensing basis for the DAEC and NRC approved methods.

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

3. The proposed amendment will not involve a significant reduction in a margin of safety.

In 1985, the reactor building crane was upgraded to single-failure-proof in compliance with NUREG–0554. The upgraded crane and handling system is in compliance with NUREG–0612, Sections 5.1.1 and 5.1.6. The NRC in NUREG–0612, Section 5.2 documented their review of the potential consequences of a load drop when handled by a single-failure-proof crane using single-failure-proof rigging compared with other alternatives and concluded as follows: ''The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after implementation of the guidelines of Section 5.1 is very low.''

This means that a load drop is considered to be unlikely within regulatory accepted standards when the load is handled by a single-failure-proof crane and handling system, and performed in accordance with Section 5.1 of NUREG–0612. A single-failure-proof crane design incorporates the applicable design basis event that in this case is a seismic event. A load drop is of such low probability that it is considered unlikely when it is handled with the reactor building crane since the crane and its handling systems satisfy the NUREG–0612 criteria for a single-failure-proof crane. Therefore, any load lifted over the spent fuel pool using the reactor building crane has a very low probability of falling into the spent fuel pool accidentally or as a result of a design basis event.

Therefore, this proposed amendment will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. Alvin Gutterman, Morgan Lewis, 1111 Pennsylvania Avenue NW., Washington, DC 20004.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: January 29, 2003.

Description of amendment request: The proposed amendment would change the drywell leakage and sump monitoring detection section of the Technical Specifications (TSs). These proposed changes clarify the definitions and restructure the coolant leakage section of the TSs and revise unidentified leakage and total leakage requirements. The revisions add a TS Limiting Condition for Operation for leakage-detection instrumentation being inoperable. This request supercedes the Nuclear Management Company's license amendment request of October 8, 2002, as supplemented November 8, 2002, which was previously noticed in the **Federal Register** on October 17, 2002 (67 FR 64144).

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

The proposed Technical Specification changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. Additionally, the proposed changes do not affect any accident previously evaluated in the Monticello Updated Safety Analysis Report (USAR). The changes simply redefine the parameters for evaluation of leakage in the drywell. The evaluation criteria for drywell leakage have been refocused into the areas that are most susceptible to IGSCC [intergranular stress corrosion cracking]. Consequently, the probability of an accident previously evaluated is not significantly increased.

The equipment referenced in the proposed changes is still required to monitor the reactor coolant system operational leakage to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is impaired. As a result, operation of the facility with the proposed changes will continue to meet the licensing basis and applicable guidelines. As such, the consequences of any accident previously evaluated are not significantly affected.

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

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

The proposed changes do not involve physical alterations of the plant; no new or different type of equipment will be installed; nor are there significant changes in the methods governing normal plant operation. The changes simply redefine the parameters for evaluation of leakage in the drywell. The evaluation criteria for drywell leakage have been refocused into the areas that are most susceptible to IGSCC. Additionally, the changes do not create any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

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

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed amendment redefines the parameters for evaluation of leakage in the drywell. There are no physical alterations of the plant; no new or different type of equipment will be installed; nor are there significant changes in the methods governing normal plant operation. Additionally, the proposed changes do not exceed or alter a design basis or safety limit as established in the Monticello licensing basis.

Therefore, these proposed changes will 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: Jay E. Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: February 11, 2003.

Description of amendment request: The proposed amendments would revise technical specification (TS) 5.5.9, ''Ventilation Filter Testing Program (VFTP)'' by (1) incorporating filter test face velocity limits for the control room special ventilation system, auxiliary building special ventilation system, spent fuel pool special and inservice purge ventilation system, and shield building ventilation system; and (2) making editorial changes. The proposed amendments would also delete the additional conditions in Appendix B of the Operating Licenses which require the licensee to complete an evaluation of the maximum test face velocity for the ventilation systems in TS 5.5.9. The additional conditions also require the licensee to submit a license amendment request for a TS amendment to specify the maximum test face velocity if the maximum actual face velocity is the greater than 110 percent of 40 fpm. Additionally, the proposed amendments would revise the penetration and system bypass limit from 0.05 percent to 0.5 percent for the ventilation systems.

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:

Revision of the Allowable Filtration Penetration and System Bypass

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

This license amendment request proposes to increase the penetration and system bypass limit for the control room special ventilation system, auxiliary building special ventilation system, spent fuel pool special and inservice purge ventilation system and shield building ventilation system from 0.05% to 0.5%. These ventilation systems are included in the plant design to mitigate accident consequences and are not assumed accident initiators, thus, this change does not involve a significant increase in the probability of an accident. This change will assure that the subject ventilation systems will perform within their intended design ranges thus, this change assures that the consequences of an accident are not increased.

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

This proposed change does not alter the design, function, or operation of any plant component and does not install any new or different equipment. The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed Technical Specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

This license amendment request proposes to increase the penetration and system bypass limit for the control room special ventilation system, auxiliary building special ventilation system, spent fuel pool special and inservice purge ventilation system and shield building ventilation system from 0.05% to 0.5%. Site dose analyses are required to demonstrate that regulatory dose limits are met using Technical Specification allowed penetration and system bypass with an appropriate safety factor as an input to the evaluation. Since the dose analyses have not been modified to credit 0.05% penetration and system bypass, this proposed change has no effect on the dose analyses which demonstrate that the regulatory limits are satisfied. Since the NRC regulatory limits must continue to be met and the safety factor will not be changed by this proposed Technical Specification change, this change does not involve a significant reduction in the margin of safety.

Addition of Filter Test Face Velocities

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

This license amendment request proposes to add filter test face velocity minimum values for the control room special ventilation system, auxiliary building special ventilation system, spent fuel pool special and inservice purge ventilation system and shield building ventilation system. These ventilation systems are included in the plant design to mitigate accident consequences and are not assumed accident initiators, thus, this change does not involve a significant increase in the probability of an accident. This change will assure that the subject ventilation systems will perform within their intended design ranges thus, this change assures that the consequences of an accident are not increased.

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

This proposed change does not alter the design, function, or operation of any plant component and does not install any new or different equipment. The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed Technical Specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

This license amendment request proposes to add filter test face velocity minimum values for the control room special ventilation system, auxiliary building special ventilation system, spent fuel pool special and inservice purge ventilation system and shield building ventilation system. These additional Technical Specification limits on system performance assures these ventilation systems are tested and maintained within their designed function limits and may increase the margin of safety for these systems. Therefore this change does not involve a significant reduction in the margin of safety.

Editorial and Administrative Changes

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

This license amendment request proposes editorial changes to Technical Specification Section 5.5.9, including replacement of ventilation system names with abbreviations and miscellaneous changes associated with addition of a new paragraph to this section, and proposes an administrative change to delete the Operating License Additional Condition for each unit that relates to NRC Generic Letter 99–02. Since these changes are editorial or administrative, they do not change any plant operating limits or technical requirements. Therefore these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not alter the design, function, or operation of any plant component and does not install any new or different equipment. The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore, the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

This license amendment request proposes editorial changes to Technical Specification Section 5.5.9, including replacement of ventilation system names with abbreviations and miscellaneous changes associated with addition of a new paragraph to this section, and proposes an administrative change to delete the Operating License Additional Condition for each unit that relates to NRC Generic Letter 99–02. Since these changes are editorial or administrative, they do not change any plant operating limits or technical requirements. Therefore these changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 11, 2003.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.1.4, ''Rod Group Alignment Limits,'' and TS 3.1.7, ''Rod Position Indication,'' to allow up to 1 hour of soak time following substantial rod movement during which individual rod position indicators may not be within its limits.

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

This license amendment request proposes to allow up to one hour of soak time following substantial rod movement during which time the rod position indication may be outside its limits. This would allow an additional hour for rod position indication to be inoperable or a control rod to be misaligned prior to entry into a Technical Specification LCO [Limiting Condition for Operation] Condition and Required Actions.

Rod position indication instrumentation is not an assumed accident initiator and thus this change does not involve a significant increase in the probability of an accident. Rod position indication instrumentation provides information on control rod position. Inoperable rod position indication instrumentation for an additional hour does not make a rod misaligned. The consequences of a rod misaligned for an additional hour are considered separately, thus inoperable rod position indication instrumentation, by itself, for an additional hour does not involve an increase in the consequences of an accident.

This license amendment request may allow a misaligned rod to be undetected for an additional hour. Plant safety analyses consider two types of rod misalignment events, static misalignment and a dropped rod. This license amendment request does not involve a significant increase in the probability of a misaligned control rod event because the one-hour time extension does not affect the control rod drive system features, whose failure would result in either type of misalignment. This proposed one-hour time extension for a control rod to be misaligned does not involve a significant increase in the consequences of a rod misalignment event as follows. The analyses show that a single dropped rod event, without any operator intervention, does not result in any fuel pin failure, therefore the rod drop event is not time dependent and an additional hour with the misalignment undetected and unmitigated does not increase the consequences of the event. Multiple rod drop events cause the reactor to trip and therefore an additional hour would not have any impact on this event.

In the static misalignment event, one or more control rods are assumed to be statically misplaced from the allowed position. This situation might occur if a rod were left behind when inserting or withdrawing banks, or if a single rod were to be withdrawn. The analysis of this event is bounded by modeling the most limiting configuration which is the control banks at the full power insertion limit except for a single control rod fully withdrawn. The analyses show that, without any operator intervention, a single fully withdrawn rod event does not result in any fuel pin failure, therefore the static rod misalignment event is not time dependent and an additional hour with the misalignment undetected and unmitigated does not increase the consequences of the event. Multiple rod misalignment events are bounded by the single rod misalignment analyses and therefore an additional hour would not have any impact on this event.

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

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

This proposed change does not alter the design, function, or operation of any plant component and does not install any new or different equipment. The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

This license amendment request proposes to allow up to one hour of soak time following substantial rod movement during which time the rod position indication may be outside its limits. This would allow an additional hour for rod position indication instrumentation to be inoperable or a control rod to be misaligned prior to entry into a Technical Specification LCO Condition and Required Actions.

The rod position indication system is an instrumentation system that provides indication to the operators that a control rod may be misaligned. Inoperable individual rod position indication instrumentation does not by itself in any way harm or impact reactor operation. Inoperable rod position indication instrumentation may impair the ability of the operators to detect a misaligned rod. The impact of inoperable rod position indication instrumentation may be offset by availability of other indications that a rod is misaligned such as nuclear instrumentation indication that reactor power has shifted to one side of the core or thermocouple indication that the core temperatures increased in one region of the core and/or decreased in another region of the core.

The Prairie Island staff is not aware of a misaligned control rod in more than 50 reactor-years of plant operation. The likelihood of a misaligned rod at Prairie Island is small and the likelihood of a misaligned rod coincident with inoperable rod position indication during the allowed one-hour extension is smaller.

The addition of one hour soak time for the rod position indication instrumentation will allow the operators and engineers to focus on monitoring the reactor performance without unnecessary entry into LCO Conditions and Required Actions with the concomitant administrative activities. Thus, these changes may enhance plant safety and reliability of equipment.

In conclusion, the proposed addition of an LCO Note in LCO 3.1.4 and 3.1.7 does not involve a significant reduction in the margin of safety because rod position indication instrumentation inoperability by itself does not impact plant safety, rod misalignment is unlikely, there may be other indications of rod misalignment, rod misalignment coincident with rod position indication instrumentation inoperability within the one hour extension is unlikely, and plant safety may be enhanced by avoiding unnecessary LCO Condition entry.

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: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 19, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) 5.3, ''Plant Staff Qualifications.'' The proposed amendments would revise requirements that have been superseded based on licensed operator training programs being accredited by the National Academy for Nuclear Training (NANT) and promulgation of the revised 10 CFR part 55, ''Operators' Licenses,'' which became effective on May 26, 1987.

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

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

Response: No.

The proposed Technical Specification change is an administrative change to clarify the current requirements for licensed operator qualifications and the licensed operator training program. With this change, the Technical Specifications continue to meet the current requirements of 10 CFR [Part] 55.

Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR [Part] 55 rule, concluded that this impact remains acceptable as long as the licensed operator training programs are certified to be accredited and are based on a systems approach to training. The Prairie Island Nuclear Generating Plant licensed operator training program is accredited by the National Academy for Nuclear Training and is based on a systems approach to training. The proposed Technical Specification change takes credit for the National Academy for Nuclear Training accreditation of the licensed operator training program. The Technical Specification requirements for all other plant staff qualifications remain unchanged.

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

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

Response: No.

The proposed Technical Specification change is an administrative change to clarify the current requirements for licensed operator qualifications and the licensed operator training program and to conform to the revised 10 CFR [Part] 55.

As discussed above, although licensed operator qualifications and training may have an indirect impact on the possibility of a new or different kind of accident from any accident previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised rule, concluded that this impact remains acceptable as long as licensed operator training programs are certified to be accredited and based on a systems approach to training. As previously noted, the Prairie Island Nuclear Generating Plant licensed operator training program is accredited by the National Academy for Nuclear Training and is based on a systems approach to training. The proposed Technical

Specification change takes credit for the National Academy for Nuclear Training accreditation of the licensed operator training program. The Technical Specification requirements for all other plant staff qualifications remain unchanged.

Additionally, the proposed Technical Specification change does not affect plant design, hardware, system operation, or procedures. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed Technical Specification change is an administrative change to clarify the current requirements applicable to licensed operator qualifications and the licensed operator training program. With this change the Technical Specifications continue to be consistent with the requirements of 10 CFR [Part] 55. The Technical Specification qualification requirements for all other plant staff remain unchanged.

Licensed operator qualifications and training can have an indirect impact on a margin of safety. However, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR [Part] 55, determined that this impact remains acceptable when licensees maintain a licensed operator training program that is accredited and based on a systems approach to training. As noted previously, the Prairie Island Nuclear Generating Plant licensed operator training program is accredited by the National Academy for Nuclear Training and is based on a systems approach to training.

The NRC has concluded, as stated in NUREG–1262, ''Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses,'' that the standards and guidelines applied by the Institute for Nuclear Power Operations' National Academy for Nuclear Training in their training accreditation program are equivalent to those put forth or endorsed by the NRC. As a result, maintaining a National Academy for Nuclear Training accredited, systems approach based licensed operator training program is equivalent to maintaining an NRC approved licensed operator training program which conforms with applicable NRC Regulatory Guides or NRC endorsed industry standards. The margin of safety is maintained by virtue of maintaining the National Academy for Nuclear Training accredited licensed operator training program.

In addition, the NRC published NRC Regulatory Issue Summary 2001–01, ''Eligibility of Operator License Applicants,'' dated January 18, 2001, ''to familiarize addressees with the NRC's current guidelines for the qualification and training of reactor operator (RO) and senior operator (SO) license applicants.'' This document again acknowledges that the Institute for Nuclear Power Operations' National Academy for Nuclear Training guidelines for education and experience, outline acceptable methods for implementing the NRC's regulations in this area.

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

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: February 6, 2003.

Description of amendment requests: The proposed license amendments would revise Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 of TS 3.3.1, ''Reactor Trip System Instrumentation,'' of the Diablo Canyon Technical Specifications. The change to SR 3.3.1.2 is consistent with NRC-approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–371. The change to SR 3.3.1.3 is editorial in nature.

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

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

The proposed change to Technical Specifications (TS) Surveillance Requirement (SR) 3.3.1.2 and SR 3.3.1.3 is consistent with the NRC approved Industry/Technical Specifications Task Force Standard Technical Specification Change Traveler, TSTF–371, and NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 2.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The reactor trip system (RTS) instrumentation will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) are not adversely affected because the change to the nuclear instrumentation system (NIS) power range channel daily surveillance assures the conservative response of the channel even at part-power levels.

The proposed change modifies the NIS power range channel daily surveillance requirement to help assure the NIS power range functions are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the USAR.

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

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

There is no hardware change or change in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected. The NIS power range high trip setpoint adjustment requirements, prior to adjusting indicated power in a decreasing power direction, will ensure the reactor power level is consistent with assumptions made in the safety analysis and licensing basis.

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

This amendment does not alter the design or performance of the Eagle 21 System, NIS, or Solid State Protection System used in the plant protection systems.

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

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

The proposed change requires a revision to the criteria for implementation of NIS power range channel adjustments based on secondary power calorimetric calculations; however, the change does not eliminate any RTS surveillances or alter the frequency of surveillances required by the Technical Specifications. The revision to the criteria for implementation of the daily surveillance will have a conservative effect on the performance of the NIS power range channels, particularly at part-power conditions. The nominal trip setpoints specified in the Technical Specification Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (FQ), nuclear enthalpy rise hot channel factor (FDH), loss of coolant accident peak cladding temperature, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of appropriate surveillance testing requirements will not reduce any margin of safety since the change will assure that safety analysis assumptions on reactor power are verified on a periodic frequency.

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

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: February 28, 2003.

Description of amendment requests: The proposed license amendments would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' to add Surveillance Requirement (SR) 3.3.1.16 to function 3.a, ''Power Range Neutron Flux Rate-High Positive Rate Trip,'' in Table 3.3.1–1. The amendments would also eliminate periodic pressure sensor response time testing (RTT) and periodic protection channel RTT.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes.

The design of the Reactor Trip System (RTS) instrumentation, specifically the positive flux rate trip (PFRT) function, will be unaffected. The reactor protection system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change imposes additional surveillance requirements to assure safety-related structures, systems, and components are verified to be consistent with the safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT function.

The Technical Specification Bases changes do not result in a condition where the design, material, or construction standards that were applicable prior to change are altered. The same RTS and engineered safety features actuation system instrumentation is being used; the time response allocations/modeling assumptions in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed change will not change any system interface and could not increase the likelihood of an accident since these events are independent of this change.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance requirements for the PFRT function. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

This change does not alter the performance of the process protection racks, nuclear instrumentation, and logic systems used in the plant protection systems. These systems will still have their response time verified by test before being placed in operational service. Changing the method of verifying instrument response for these systems (assuring equipment operability) from time response testing to channel and calibration checks will not create any new [accident] initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristic to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by periodic surveillance.

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

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

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor, nuclear enthalpy rise hot channel factor, loss of coolant accident peak cladding temperature, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis are changed. The imposition of additional surveillance requirements maintains the margin of safety by assuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for the process protection racks, nuclear instrumentation, and logic systems are modified to allow use of engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will continue to be performed and may be used to detect any degradation which might cause the response time to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant 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: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

Pacific Gas and Electric Company, Docket No. 50–323, Diablo Canyon Nuclear Power Plant, Unit No. 2, San Luis Obispo County, California

Date of amendment request: March 3, 2003.

Description of amendment request: The proposed license amendment would revise Technical Specification (TS) 5.5.9, ''Steam Generator Tube Surveillance Program,'' and TS 5.6.10, ''Steam Generator Tube Inspection Report,'' for Diablo Canyon Power Plant (DCPP) Unit 2, to apply a probability of detection (POD) of 1.0 to the bobbin indication in the steam generator (SG) 4 tube at row 44, column 45 at the second tube support plate (TSP) on the hot leg side (R44C45–2H) for the beginning of cycle (BOC) voltage distribution for the DCPP Unit 2 BOC Cycle 12 operational assessment.

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 use of probability of detection (POD) of 1.0 for the bobbin indication in the Diablo Canyon Power Plant (DCPP) Unit 2 steam generator (SG) 4 tube at row 44, column 45 at the second tube support plate (TSP) on the hot leg side (R44C45–2H) for the beginning of cycle (BOC) voltage distribution for the DCPP Unit 2 BOC cycle 12 operational assessment does not increase the probability of an accident. Based on industry and plant specific bobbin detection data for outside diameter stress corrosion cracks (ODSCC) within the SG tube support plate region, large voltage bobbin indications, such as those the size of indication R44C45–2H, can be detected with 100 percent certainty. Since large voltage ODSCC bobbin indications within the SG TSP can be detected, they will not be left in service, and therefore these indications should not be included in the voltage distribution for the purpose of operational assessments. Therefore, these large voltage indications will not result in an increase in the probability of a steam generator tube rupture (SGTR) accident or an increase in the consequences of a SGTR or main steam line break (MSLB) accident.

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

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

The use of a POD of 1.0 for the DCPP Unit 2 R44C45–2H bobbin indication for the BOC voltage distribution for the DCPP Unit 2 BOC cycle 12 operational assessment concerns the SG tubes and can only affect the SGTR accident. Since the SGTR accident is already considered in the Final Safety Analysis Report Update, there in [is] no possibility to create a design basis accident which has not been previously evaluated.

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

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

The use of POD of 1.0 for the DCPP Unit 2 R44C45–2H bobbin indication for the BOC voltage distribution for the DCPP Unit 2 BOC cycle 12 operational assessment does not involve a significant reduction in a margin of safety. The applicable margin of safety potentially impacted is the Technical Specification 5.6.10, ''Steam Generator Tube Inspection Report,'' projected end-of-cycle leakage for a MSLB accident and the projected end-of-cycle probability of burst. Based on industry and plant specific bobbin detection data for ODSCC within the SG tube support plate region, large voltage bobbin indications, such as those the size of indication R44C45–2H, can be detected with 100 percent certainty and will not be left in service. Therefore these indications should not be included in the voltage distribution for the purpose of operational assessments. Therefore, these large voltage indications will not result in a significant increase in the actual end-of-cycle leakage for a MSLB accident or the actual end-of-cycle probability of burst.

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

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: February 14, 2003.

Description of amendment request: The proposed amendment would extend the surveillance test intervals and allowed out-of-service times for the end-of-cycle recirculation pump trip instrumentation.

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

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

Response: No.

The proposed amendment would extend the allowed out-of-service times (AOTs) and surveillance test intervals (STIs) for the end of cycle recirculation pump trip (EOC-RPT) instrumentation system. No changes are being made to any EOC-RPT instrumentation setpoints or components. The effect of the proposed changes is to reduce the potential for unnecessary plant scrams or transients. The proposed changes were evaluated in General Electric Company Topical Report GENE–770–06–1–A which concluded that they do not result in a degradation in overall plant safety.

Since the proposed changes do not affect any accident initiator, and since the EOC-RPT instrumentation will remain capable of performing its design function, 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.

Extending the AOTs and STIs for the EOC-RPT instrumentation does not change the design function or operation of any plant equipment. Additionally, no new modes of plant operation are involved with these changes.

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.

No changes are being made to any plant instrumentation setpoints or to the required level of redundancy. The proposed changes were evaluated in General Electric Company Topical Report GENE–770–06–1–A, which concluded that they do not result in a degradation in overall plant safety.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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: January 29, 2003.

Description of amendment request: The licensee proposed administrative and editorial changes to the Salem Nuclear Generating Station (Salem), Unit No. 1 and Unit No. 2 Technical Specifications (TSs) as follows: (1) The second equation in Salem Unit No. 2 TS

Limiting Condition for Operation 3.2.2 on page 3/4 2–5 will be revised; (2) Salem Unit No. 2 TS Table 3.3–6 will be revised to indicate that one operable channel of containment air particulate activity reactor coolant system (RCS) leakage detection instrumentation is required for operation in Modes 1 through 4; (3) Salem Unit No. 1 TS 3/4.7.6 Action Statements ''d.'' (for Modes 1, 2, 3 and 4) and ''e.'' (for Modes 5 and 6) will be revised to refer to Action 25 in TS Table 3.3–6; and (4) Salem Unit No. 2 TS 3/4.7.6 Action Statements ''d.'' (for Modes 1, 2, 3 and 4) and ''e.'' (for Modes 5 and 6) will be revised to refer to Action 28 in TS Table 3.3–6.

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

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

Response: No.

The proposed changes to the TSs are administrative or editorial in nature and do not change the intent of any Technical Specification requirement. No changes are being made to any plant systems, structures or components (SSCs).

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 administrative and editorial changes to the TSs do not change the design function or operation of any plant equipment. Additionally, no new modes of plant operation are involved with these changes.

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 proposed changes are administrative and editorial corrections to the TSs that do not affect the ability of plant SSCs to perform their design basis accident functions. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 11, 2003.

Description of amendment request: The amendment application requests a revision to the Unit 1 defueled Technical Specifications administrative controls section to propose changes in organizational responsibilities. Specifically, the proposed change identifies that the Vice President, Engineering & Technical Services would be responsible for decommissioning activities. Additionally, the Station Manager would be designated as having approval authority for activities within the Station Manager's organization.

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

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

No. This is a request to revise the San Onofre Nuclear Generating Station, Unit 1 permanently defueled technical specifications administrative controls. The proposed administrative changes are due to a realignment of the Unit 1 Decommissioning Project into the Engineering & Technical Services organization and the establishment of the Station Manager position within the Nuclear Generation organization. The proposed changes identify the Vice President, Engineering & Technical Services to be responsible for decommissioning activities and provides the Station Manager the opportunity to approve procedures and changes to procedures and changes to the Process Control Program that are under the Station Manger's responsibility. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes are administrative. Therefore, the proposed changes do not involve the possibility of a new or different type of accident from any accident previously evaluated.

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

No. The proposed changes are administrative. Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis. These administrative changes do not affect the design or operation of the facility and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Acting Section Chief: Mark Thaggard.

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

Date of amendments request: March 25, 2003.

Description of amendments request: The proposed amendments would revise Technical Specification 3.5.2, ''ECCS—Operating,'' Surveillance Requirement (SR) 3.5.2.5. Specifically, the proposed change would replace the requirement to verify specific surveillance test values for the Emergency Core Cooling System (ECCS) pumps with the requirement to verify the developed head for each ECCS pump in accordance with the Inservice Testing Program. This new requirement is identical to SR 3.5.2.4 in NUREG–1432, ''Standard Technical Specifications, Combustion Engineering Plants,'' Revision 2.

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

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

Response: No.

Deleting the specific surveillance test values for Emergency Core Cooling System (ECCS) pumps from Surveillance Requirement (SR) 3.5.2.5 does not affect the probability of occurrence or consequences of an accident previously evaluated because ECCS pumps are for accident mitigation and do not contribute to initiation of accidents. Periodic surveillance testing of the ECCS pumps in accordance with the Inservice Testing (IST) program provides assurance that the pumps will perform as assumed in the safety analysis. There is no change to the safety analysis.

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.

ECCS pumps are for accident mitigation and do not contribute to accident initiation. The ECCS system will still be verified capable of meeting its emergency core cooling and IST requirements. There is no change to the safety analysis.

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

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

Response: No.

There is no change to the safety analysis. Testing of the ECCS pumps as required by the IST Program combined with the existing Technical Specification 3.5.2—''ECCS—Operating'' surveillance requirements ensure that the ECCS requirements remain met without a significant reduction in a margin of safety. Therefore, there is no significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: March 13, 2003.

Description of amendment request: The proposed amendments would modify the Sequoyah Nuclear Plant, Units 1 and 2, Operating Licenses DPR–77 and DPR–79. This proposed request provides Technical Specification (TS) change 03–01 that would revise the limiting condition for operation for TS Section 3.5.1, ''Cold Leg Injection Accumulators'' and TS Section 3.5.5, ''Refueling Water Storage Tank.'' This revision would modify the single boron concentration requirement by inserting a table that defines the minimum and maximum amount of boron that is required for accident mitigation based on the number of tritium producing rods in the core.

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?

The proposed change modifies the required boron concentration for the cold leg accumulators (CLAs) and refueling water storage tank (RWST). The proposed values have been verified to maintain the required accident mitigation safety function for the CLAs and RWST. The CLAs and RWST safety function is to mitigate accidents that require the injection of borated water to cool the core and to control reactivity. These functions are not potential sources for accident generation and the modification of the boron concentration that supports event mitigation will not increase the potential for an accident. Therefore, the possibility of an accident is not increased by the proposed changes. The boron levels for this change are based on the number or tritium producing rods in the core. As the number of rods is increased the need for additional shutdown boron also increases. This effect has been evaluated with the same methodology utilized for previous NRC approved amendments associated with tritium production. This methodology ensures that the impact of tritium producing rods is adequately compensated for by the required boron concentrations and has been incorporated into the proposed revision. Since the boron levels will continue to maintain the safety function of the CLAs and RWST in the same manner as currently approved, the consequences of an accident is not increased by the proposed changes.

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

The proposed change only modifies boron concentrations for accident mitigation functions of the CLAs and RWST. These functions do not have a potential to generate accidents as they only serve to perform mitigation functions associated with an accident. The proposed requirements will maintain the mitigation function in an identical manner as currently approved. There are no plant equipment or operational changes associated with the proposed revision other than the adjustment of the boron level in the CLAs and RWST. Therefore, since the CLA and RWST functions are not altered and the plant will continue to operate without change, the possibility of a new or different kind of an accident is not created.

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

This change proposes boron concentration requirements that support the accident mitigation functions of the CLAs and RWST equivalent to the currently approved limits. The proposed change does not alter any plant equipment or components and does not alter any setpoints utilized for the actuation of accident mitigation system or control functions. The proposed boron values have been verified to provide the same level of reactivity control for accident mitigation. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: March 12, 2003.

Description of amendment request: The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) and the Technical Specification Bases. The revision would update the quality assurance criteria and the basis for the seismic qualification of the ducting installed as part of the suspended ceiling air delivery system in the main control room (MCR).

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 design function of the MCR ducting system is to support pressurization and cooling of the control room during normal and accident conditions. The MCR ducting is a passive plant feature and does not act as an accident initiator. Consequently, the changes in the MCR ducting system and suspended ceiling quality assurance (QA) requirements and qualification methodology do not result in an increase in the probability of an accident previously evaluated.

For the principal design basis accidents, Loss of Coolant Accident (LOCA), Internal Flood, Steam Generator Tube Rupture (STGR), Main Steam Line Break (MSLB), etc., the integrity of the MCR HVAC [heating, ventilation, and air conditioning] system, including the suspended ceiling, will not be compromised. These accidents do not have a structural effect on the MCR. This means that for postulated radiological or toxic chemical accidents, the ability to both pressurize and maintain MCR temperatures within the design limits is unaffected by the limited QA and newly defined seismic requirements for the air delivery components.

An accident that involves a fire that affects the MCR or the habitability of the MCR was not a consideration for the qualification of the air distribution components. A fire of this nature will result in plant operation from the Auxiliary Control Room which is supported by a separate heating, ventilation and air conditioning (HVAC) supply system.

An earthquake (including the Design Basis SSE [safe shutdown earthquake]) is the only event for which the design basis for the MCR HVAC and suspended ceiling is potentially challenged. A seismic qualification report by an industry seismic expert concludes that the air delivery components will remain in place, will retain their structural integrity such that flow will not be impeded, and the pressure boundary will not be lost during and subsequent to a design basis seismic event. Further, as assured by TVA's qualification report, the suspended ceiling will remain in place during and subsequent to a seismic event or accident. Thus, the revised QA and seismic qualification requirements for the MCR air delivery components and suspended ceiling will not result in loss of safety function for any design basis accident or event. Consequently, the accident dose consequences as previously evaluated in the UFSAR are not affected by the proposed license amendment.

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

No. The MCR air delivery components addressed by the proposed amendment are not an accident initiator and therefore, failure of these components will not initiate a design basis accident. In addition, the subject air delivery components and suspended ceiling have been seismically qualified, as previously discussed, and a determination has been made that they will not fail during a design basis accident. Therefore, the air delivery components and suspended ceiling will continue to perform their safety function during normal and accident conditions. Consequently, this activity does not create a possibility of a new or different type of accident than any previously evaluated.

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

No. The changes addressed in TVA's proposed amendment are associated with changes in QA requirements and seismic qualification methodology for safety related air delivery components and for the suspended ceiling. The change does not affect specific HVAC equipment safety limits, design limits, set points, or other critical parameters. In addition, the new seismic analysis methodology and limited QA requirements ensure that these components will continue to perform their safety functions during normal and accident conditions. The previously implied margin of safety against structural or functional failure of the air delivery components or suspended ceiling during and after a design basis SSE has not been reduced. Consequently, the MCR HVAC system or suspended ceiling margin of safety has not been significantly reduced by this proposed amendment.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: March 24, 2003.

Description of amendment request: The proposed amendment would revise the design and licensing basis failure modes and effects analysis for specific valves in the essential raw cooling water system, component cooling water system, and control air system. Tennessee Valley Authority has identified a condition where containment integrity, accident flood levels, and sump boron concentrations subsequent to a high-energy line break events could not be assured automatically as stated in the updated final safety analysis report (UFSAR). In certain postulated events, manual actions may be required using equipment not currently evaluated in the UFSAR.

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

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

Response: No.

The manual actions required by this change are only needed after a high energy line break (HELB) accident, such as a loss-of-coolant-accident (LOCA), main steam line break (MSLB), feedwater line break accidents, etc., has occurred inside containment and a single failure of an outboard containment isolation valve to close has occurred on one of four specific lines inside containment. In this event, the manual actions ensure containment is isolated, which is consistent with the current design. Consequently, the manual actions of isolating the air and water lines after an accident do not affect the frequency of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

The UFSAR currently indicates that the containment vessel design and the containment isolation system automatically ensure containment integrity is maintained and thus ensure that release of radioactive material from containment remains below allowable limits during and subsequent to an accident. Current UFSAR Failure Modes and Effects Analysis (FMEA) for the affected essential raw cooling water (ERCW), component cooling system (CCS), and control air system (CAS) valves indicate a single failure of the outboard containment isolation valve in conjunction with a concurrent accident and consequential (due to interaction) failure of the system piping inside containment, has no adverse effect on the plant; thus, containment integrity is ensured automatically. This change revises these evaluations to indicate manual actions are required to ensure containment integrity in the event of an HELB and single failure of an outboard containment isolation valve. Evaluations have been performed to ensure adequate instrumentation and time is available to recognize the need and to manually isolate an affected line subsequent to an HELB if the outboard containment isolation valve does not close. The emergency procedures have been revised that require manual actions to be performed to isolate CAS, ERCW, and CCS and to open and close a post accident sampling facility (PASF) cooling water supply valve. The Operations Staff has confirmed that the subject containment lines can be isolated within the allowable time and without exceeding the dose limitations as required by 10 CFR [Part] 50, Appendix A, General Design Criteria (GDC) 19, ''Control Room.''

Evaluations have indicated that adequate instrumentation, time, and staffing are available to manually isolate the lines into containment. Operator actions are achievable and can be accomplished without heroic actions. Therefore, containment integrity from overpressurization or flooding is maintained within the current design basis analysis, and the radiological consequences of an accident will not be increased by this change. Consequently, 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.

This change implements manual actions to isolate four specific containment lines in lieu of automatic containment isolation for previously identified accidents. The manual actions are required to maintain containment integrity from overpressurization, containment flood levels, sump pH levels, and emergency core cooling system (ECCS) water boron concentrations subsequent to an HELB inside containment concurrent with a single failure of an outboard containment isolation valve on a CAS, ERCW, or CCS line. The UFSAR FMEA evaluations will be revised by this proposed change to include the failure modes and associated manual actions.

NRC Information Notice (IN) 97–78, ''Crediting of Operator Actions in Place of Automatic Actions and Modifications of

Operator Actions, including Response Times,'' provided guidance to the industry concerning use of operator actions in place of automated system or component actuation. IN 97–78 states: In those instances where licensees consider temporary or permanent changes to the facility which credit operator actions, the NRC has relied on the guidance provided in * * * ANSI/ANS 58.8, ''Time Response Design Criteria for Safety-related Operator Actions,'' * * * for evaluating such changes. The American Nuclear Society (ANS)–58.8, establishes the requirements for safety-related operator actions, which are summarized as follows: (1) The specific operator actions required, (2) the potentially harsh or inhospitable environmental conditions expected, (3) ingress/egress paths taken by the operators to accomplish functions, (4) procedural guidance for required actions, (5) operator training and qualifications to carry out actions, (6) any additional support personnel and/or equipment to carry out actions, (7) information required by the control room staff to determine whether action is required, including qualified instrumentation to diagnose the situation and to verify that the action is successfully, (8) ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery, and (9) consideration of risk significance of operator actions.

The manual actions implemented by this change can be completed within the guidance and criteria provided in IN 97–78 and ANS–58.8. Consequently, the manual actions can be credited in the mitigation of the specific accidents. With credit for the manual actions to isolate the affected lines subsequent to an accident inside containment, the type of accidents and consequences currently evaluated in the UFSAR, remains the same. Therefore, the proposed change does not create the possibility of new or different kinds of accidents from any accident previously evaluated.

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

Response: No.

This change establishes requirements for manual actions to isolate one air line and three water lines subsequent to an accident inside containment concurrent with a single failure of a containment isolation valve to close. The manual actions ensure air or water cannot continue to enter containment with a single failure of an outboard containment isolation valve when the line pressure boundary inside containment is lost due to an accident and associated pipe interactions. The safety-related configuration of the lines (outboard motor operated valve and inboard check valve) continues to ensure the containment environment is automatically prevented from exiting the line to outside the containment. Safety-related instrumentation is available to inform operators that the manual actions are required, and operators have been trained in the requirements for addressing the failures of valves to close. In addition, adequate time and resources are available to perform the manual actions. The manual actions meet the criteria for safety-related operator actions contained in NRC IN 97–78 and ANS–58.8. Further, the proposed change to allow credit for the manual actions does not affect the offsite and Main Control Room dose consequences of the accidents currently reported in UFSAR Chapter 15, Accident Analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: March 6, 2003.

Brief description of amendments: Technical Specifications Section 1.1 ''Definitions'' for Engineered Safety Feature (ESF) Response Time and Reactor Trip System (RTS) Response Time require U.S. Nuclear Regulatory Commission (NRC) review and approval of any methodology used to allocate response times in lieu of measuring them. The application requests NRC review and approval of a topical report to allow the use of allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time.

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

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

Response: No.

The proposed change does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS and ESFAS [Engineered Safety Feature Actuation System] instrumentation are being used and the time response allocations and modeling assumptions in the Chapter 15 safety analysis are unchanged. Only the method of verifying the time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade, or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR [Final Safety Analysis Report]. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not alter the performance of the process protection racks, the nuclear instrumentation, or the logic systems used in the plant protection systems. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristics to exceed the total allowance. Changing the method of periodically verifying instrument response for these systems from response time testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristic to exceed the total allowance. The total time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

The proposed change does not affect the total system response time assumed in the safety analysis. The periodic response time verification method for the Process protection racks, the nuclear instrumentation and the logic systems is modified to allow the use of actual test data or engineering data. The method of verification still provides assurance that the total system response time is within that defined in the safety analysis, since calibration tests will continue to be performed and may be used to detect any degradation which might cause the response time to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Therefore the proposed change does not involve a reduction in a margin of safety.

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 18, 2003.

Brief description of amendments: The proposed amendment would delete certain of the Surveillance Requirements in Technical Specification 3.6.3 entitled ''Containment Isolation Valves.''

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

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

Response: No.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Final Safety Analysis Report] are not adversely affected.

The proposed changes will not involve a significant increase in the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

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

Response: No.

The proposed change does not involve any physical alteration of the units. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints at which protective or mitigative actions are initiated that are affected by the proposed change. The proposed change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures, which ensure the unit remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change does not alter assumptions made in the safety analyses.

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

(3) Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

The proposed change will not adversely affect operation of plant equipment and will not result in a change to the setpoints at which protective actions are initiated. None of the acceptance criteria for any accident analysis is changed. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (FQ), nuclear enthalpy rise hot channel factor (FDH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and 2, Louisa County, Virginia

Date of amendment request: December 13, 2002.

Description of amendment request: The proposed amendments will extend the Completion Time of Technical Specification (TS) 3.8.7, Inverters-Operating, Required Action A.1, from 24 hours to 14 days for an inoperable inverter on either Train H or Train J.

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

The proposed change to extend the Completion Time for an inoperable inverter from 24 hours to 14 days does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. In addition, this proposed change will not alter assumptions relative to the mitigation of an accident or transient event.

The licensee performed an evaluation to determine the risk significance of the proposed change. This risk evaluation concluded that the increases in annual core damage frequency (CDF) and large early release frequency (LERF) associated with the proposed change can be characterized as ''very small changes'' by Regulatory Guide (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.'' Additional evaluation by the licensee determined that the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) associated with the proposed change are within the acceptance criteria in RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.'' Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed change to extend the Completion Time for an inoperable inverter has been evaluated for its effect on plant safety. The licensee's risk-informed evaluation concluded that the increases in annual CDF and LERF associated with the proposed change can be characterized as ''very small changes'' by RG 1.174. The

ICCDP and ICLERP associated with the proposed change are within the acceptance criteria in RG 1.177. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: November 27, 2002.

Brief description of amendment: This amendment deletes technical specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminates the requirements to have and maintain the post accident sampling system at the Clinton Power Station, Unit 1. The amendment also addresses related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: March 21, 2003.

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

Amendment No.: 155.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2797).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 2003.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: April 10, 2002, as supplemented February 12, 2003.

Brief description of amendment: The amendment deleted Technical Specification 4.6.1.c, related to 24-month emergency diesel generator surveillance, and relocated these requirements to the Updated Final Safety Evaluation Report (UFSAR).

Date of issuance: April 3, 2003.

Effective date: As of the date of issuance and shall be implemented within 30 days, including the relocation of the emergency diesel generator maintenance requirements of Technical Specification 4.6.1.c to the Updated Final Safety Analysis Report (UFSAR), as was described in the licensee's application dated April 10, 2002, and evaluated in the NRC staff's safety evaluation dated April 3, 2003, and which relocation shall be included in the next scheduled update of the UFSAR pursuant to 10 CFR 50.71(e).

Amendment No.: 243.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 28, 2002 (67 FR 36926).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 3, 2003.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: July 24, 2002, as supplemented February 21, 2003.

Brief Description of amendments: The amendments revise the Technical Specifications Section 3.1.7, ''Standby Liquid Control (SLC) System,'' to reflect modifications being made to the system as a result of transition to the GE14 fuel design.

Date of issuance: March 25, 2003.

Effective date: March 25, 2003.

Amendment Nos.: 227 and 255.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications and Appendix B, ''Additional Conditions.''

Date of initial notice in ***Federal Register:*** August 20, 2002 (67 FR 53984).

The February 21, 2003, supplement contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 25, 2003.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of amendment request: November 7, 2002, as supplemented February 17, 2002.

Brief description of amendment: The amendment revises the Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2 from 1.09 to 1.11

for two recirculation loop operation and from 1.10 to 1.13 for single recirculation loop operation.

Date of issuance: March 25, 2003.

Effective date: March 25, 2003.

Amendment No.: 254.

Facility Operating License No. DPR–62: Amendment changes the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 10, 2002 (67 FR 75869). The February 17, 2003, supplement contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 25, 2003.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: August 28, 2002, as supplemented November 21, 2002.

Brief description of amendment: This amendment revises the Technical Specifications (TS) by adding Topical Report EMF–2328 (P)(A), ''PWR Small Break LOCA Evaluation Model, S–RELAP5 Based'' as reference in the TS to allow the licensee to update the methodologies that are used for safety analyses for the Shearon Harris Nuclear Power Plant, Unit 1. The amendment also relocates referenced methodologies within TS 6.9.1.6.2 to group mechanical design methodologies together.

Date of issuance: March 28, 2003.

Effective date: March 28, 2003.

Amendment No.: 114.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 15, 2002 (67 FR 63691). The November 21, 2002, supplement 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 March 28, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 23, 2002, as supplemented December 20, 2002, and February 27, 2003.

Brief description of amendment: The amendment revises the Fermi 2 Technical Specifications (TSs) to allow a one-time deferral of the Type A primary containment integrated leak rate test. Specifically, TS 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' would be revised to extend the current interval for performing the containment Type A test to 15 years.

Date of issuance: March 27, 2003.

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

Amendment No.: 153.

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

*Date of initial notice in **Federal Register:*** June 25, 2002 (67 FR 42817). The supplemental letters 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 amendment is contained in a Safety Evaluation dated March 27, 2003.

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: November 6, 2001, as supplemented on December 27, 2001, and July 15, August 6, and October 29, 2002.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) associated with the spent fuel pool (SFP). Specifically, the amendment increases the allowable nominal average fuel assembly enrichment from 4.5 weight percent (w/o) Uranium-235 (U–235) to 4.85 w/o U–235 for all regions of the SFP, the new fuel storage racks (dry), and the reactor core; allows fuel to be located in the 40 storage cells in Region B of the SFP that are currently empty and blocked; credits SFP soluble boron for reactivity control during normal conditions; and reduces the Boraflex reactivity credit in Regions A and B of the SFP.

Date of issuance: April 1, 2003.

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

Amendment No.: 274.

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

*Date of initial notice in **Federal Register:*** February 19, 2002 (67 FR 7414). The supplement dated December 27, 2001, provided a revision to the licensee's analysis of the issue of no significant hazards consideration, as originally provided in the November 6, 2001, application. The supplements dated July 15, August 6, and October 29, 2002, 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 April 1, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 3, 2002, as supplemented on January 23, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.4.9, ''Pressurizer,'' to increase the pressurizer water level limit when the plant is in MODE 3 (Hot Standby). The pressurizer water level limit for MODES 1 and 2 (Power Operation and Startup) remains unchanged. The amendment also revises TS 3.8.4, ''DC Sources—Operating,'' to remove the notes that refer to the one-time amendment allowing the online replacement of station batteries 31 and 32. The notes were no longer applicable since the batteries have been replaced.

Date of issuance: March 25, 2003.

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

Amendment No.: 216.

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

*Date of initial notice in **Federal Register:*** July 9, 2002 (67 FR 45566).

The January 23 letter provided clarifying information that did not enlarge the scope of the original **Federal Register** notice or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 25, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 4, 2002, which replaces the original applications dated May 1, 2002.

Brief description of amendment: The proposed amendment would change the Pilgrim Nuclear Power Station Technical Specification (TS) Figures 3.6.1, 3.6.2, and 3.6.3 to extend the applicability of the current reactor pressure vessel pressure-temperature (P–T) curves through the end of Operating Cycle (OC) 16. The current P–T curves were approved for use in License Amendment 190, dated April 13, 2001, and are limited to use through the end of OC 14. The proposed change would delete the 20 and 32 Effective Full Power Year curves and replace the wording of the title blocks to allow use through the end of OC 16.

Date of issuance: March 28, 2003.

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

Amendment No.: 197.

Facility Operating License No. DPR–35: This amendment revised the TS.

Date of initial notice in ***Federal Register:*** February 18, 2003 (68 FR 7816).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 28, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 19, 2002, as supplemented by letter dated December 19, 2002.

Brief description of amendment: The amendment revises the Technical Specifications by: (1) Modifying the wording of the current Surveillance Requirement (SR) 4.0.1 and SR 4.0.3 to be consistent with NUREG–1431, Revision 2, Improved Standard Technical Specifications (ISTS) wording for SR 3.0.1 and SR 3.0.3; and (2) modifying the ISTS wording, adopted in Item (1), above, for SR 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of up to 24 hours ''* * * when the allowable outage time limits of the ACTION requirements are less than 24 hours'' to ''* * * up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater.'' In addition, the following requirement is added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: March 21, 2003.

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

Amendment No.: 187.

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

Date of initial notice in ***Federal Register:*** February 4, 2003 (68 FR 5670).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 20, 2002.

Brief description of amendment: This amendment approves several administrative changes to the Waterford Steam Electric Station, Unit 3 Technical Specifications (TSs) to revise, correct, or clarify certain titles, page numbers, and heading information. It also revises personnel and committee titles that have been changed, revises administrative reporting requirements to conform to 10 CFR 50.4, and deletes redundant or unnecessary requirements from TSs 5.4, 6.6, and 6.7.

Date of issuance: April 3, 2003.

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

Amendment No.: 188.

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

Date of initial notice in ***Federal Register:*** February 4, 2003 (68 FR 5673).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 16, 2002.

Brief description of amendments: The amendments modify the Unit 3 allowable value Technical Specification, and the Units 2 and 3 surveillance requirements Technical Specification for the reactor protection system scram discharge volume water level-high function.

Date of issuance: April 3, 2003.

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

Amendment Nos.: 198/191.

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

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68737).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 27, 2002.

Brief description of amendments: These amendments delete Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminate the requirements to have and maintain the post accident sampling system at the LaSalle County Station, Units 1 and 2. The amendments also address related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: March 21, 2003.

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

Amendment Nos.: 158/144.

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

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2802).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 22, 2002.

Brief description of amendments: The amendments modify the required surveillance interval from monthly to quarterly for calibration of the trip units associated with the instrumentation channels of the Anticipated Transient Without Scram-Recirculation Pump Trip system.

Date of issuance: April 1, 2003.

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

Amendment Nos.: 213 and 207.

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

Date of initial notice in ***Federal Register:*** October 1, 2002 (67 FR 61682). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 1, 2003.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendment: October 16, 2002, as supplemented January 28, 2003.

Brief description of amendment: The amendment would revise the Technical Specification values for the 4 kilovolt degraded-voltage and loss-of-voltage relays.

Date of issuance: March 26, 2003.

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

Amendment No.: 256.

Facility Operating License No. DPR–74: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68739).

The supplemental letters 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 amendment is contained in a Safety Evaluation dated March 26, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 19, 2002.

Brief description of amendment: The amendment revised the Technical Specifications to add the definition of shutdown margin (SDM), incorporate new, more restrictive SDM limits, add the associated limiting condition for operation actions and completion times for each applicable operating condition if the SDM is not met, and add surveillance requirements for verifying SDM.

Date of issuance: March 27, 2003.

Effective date: March 27, 2003.

Amendment No.: 180.

Facility Operating License No. DPR–63: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2806).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 27, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 29, 2002, as supplemented by letter dated January 24, 2003.

Brief description of amendment: The amendment changes the surveillance requirement of TS 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time 5-year extension to the 10-year interval for performing the next Type A containment integrated leakage rate test (ILRT). The change allows ILRT testing within 15 years from the last ILRT, which was performed in September 1993.

Date of issuance: March 21, 2003.

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

Amendment No.: 249.

Facility Operating License No. DPR 49: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 30, 2002 (67 FR 21291).

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 amendment is contained in a Safety Evaluation dated March 21, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 22, 2002, as supplemented October 25, 2002, January 23, and February 12, 2003.

Brief description of amendment: The amendment changes TS Surveillance Requirement 4.7.A.2.b, ''Primary Containment Integrity,'' to allow a one-time, 5-year extension to the 10-year interval for performing the next Type A containment integrated leakage rate test (ILRT). The change allows ILRT testing within 15 years from the last ILRT, which was performed in March 1993.

Date of issuance: March 31, 2003.

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

Amendment No.: 134.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 3, 2002 (67 FR 56324).

The October 25, 2002, January 23, and February 12, 2003, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 25, 2002, as supplemented on October 21, 2002.

Brief description of amendment: The amendment would modify Technical Specification (TS) requirements for missed surveillance tests in TS 4.0.3 using the Consolidated Line Item Improvement Program, modify TS 4.0.1 to be consistent with the Standard Technical Specifications (STS), and incorporate a TS Bases Control Program in Section 6.0 in accordance with the STS.

Date of issuance: March 31, 2003.

Effective date: March 31, 2003.

Amendment No.: 145.

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

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75883)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 2003.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: April 4, 2002, as supplemented by letter dated January 9, 2003.

Brief Description of amendments: The amendments revise Technical

Specifications 5.5.17, ''Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The 10-year interval between ILRTs is to be extended to 15 years from the previous ILRTs that were completed in March 1994 for Unit 1 and March 1995 for Unit 2.

Date of issuance: March 21, 2003.

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

Amendment Nos.: 159/150.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68743).

The supplement, dated January 9, 2003, provided clarifying information that did not change the scope of the April 4, 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 March 21, 2003.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of application for amendment: November 15, 2002, as supplemented by letters dated February 19, 2003, and February 26, 2003.

Description of amendment: This one-time condition establishes special provisions and requirements for safe operation of Unit 2 while heavy load lifts are performed during the Unit 1 steam generator replacement project. The provisions for heavy load lifts are described in Topical Report 24370–TR–C–002, which was previously submitted on April 15, 2002, for NRC review and approval. The topical report contains prerequisite actions for heavy load movement, active monitoring during heavy load movement, and compensatory measures in response to the unlikely event of a heavy load drop.

Date of issuance: March 26, 2003.

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

Amendment No.: 273.

Facility Operating License No. DPR–79: Amendment revises the Operating License.

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75885).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 7th day of April, 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–9026 Filed 4–14–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Model Application Concerning Technical Specifications Improvement Regarding Scram Discharge Volume Vent and Drain Valves Actions for Boiling Water Reactors Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model safety evaluation (SE), a model no significant hazards consideration (NSHC) determination, and a model license amendment application relating to a change in the technical specifications (TSs) required actions for inoperable vent and drain valves for the scram discharge volume (SDV) for boiling water reactors (BWRs). The purpose of these models is to permit the NRC to efficiently process amendments that propose to incorporate this change into plant-specific TS. Licensees of nuclear power reactors to which the models apply may request amendments utilizing the model application.

DATES: The NRC staff issued a **Federal Register** Notice (68 FR 8637, February 24, 2003) which provided a model SE and a model NSHC determination related to changing the completion times to address inoperable valves in SDV vent or drain lines. The NRC staff hereby announces that the model SE and NSHC determination may be referenced in plant-specific applications. The staff has posted a model application on the NRC web site to assist licensees in using the consolidated line item improvement process (CLIIP) to incorporate this change. The NRC staff can most efficiently consider applications based upon the model application if the application is submitted within a year of this **Federal Register** Notice.

FOR FURTHER INFORMATION CONTACT: William Reckley, Mail Stop: O–7D1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone 301–415–1323.

SUPPLEMENTARY INFORMATION:

Background

Regulatory Issue Summary 2000–06, ''Consolidated Line Item Improvement Process for Adopting Standard Technical Specifications Changes for Power Reactors,'' was issued on March 20, 2000. The CLIIP is intended to improve the efficiency of NRC licensing processes. This is accomplished by processing proposed changes to the standard technical specifications (STS) in a manner that supports subsequent license amendment applications. The CLIIP includes an opportunity for the public to comment on proposed changes to the STS following a preliminary assessment by the NRC staff and finding that the change will likely be offered for adoption by licensees. The CLIIP directs the NRC staff to evaluate any comments received for a proposed change to the STS and to either reconsider the change or to proceed with announcing the availability of the change for proposed adoption by licensees. Those licensees opting to apply for the subject change to TS are responsible for reviewing the staff's evaluation, referencing the applicable technical justifications, and providing any necessary plant-specific information. Each amendment application made in response to the notice of availability will be processed and noticed in accordance with applicable rules and NRC procedures.

This notice involves changes to required actions for inoperable SDV vent and drain valves for BWRs. This proposed change was proposed for incorporation into the STS by the BWR Owners Group as Technical Specification Task Force (TSTF)–404, Revision 0.

Applicability

This proposed change to required actions for inoperable SDV vent and drain valves is applicable to BWRs.

The CLIIP does not prevent licensees from requesting an alternative approach or proposing the changes without referencing the model SE and the NSHC. Variations from the approach recommended in this notice may, however, require additional review by the NRC staff and may increase the time and resources needed for the review.

Public Notices

In a notice in the **Federal Register** dated February 24, 2003 (68 FR 8637), the staff requested comment on the use