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 October 18, 2002, through October 31, 2002. The last biweekly notice was published on October 29, 2002 (67 FR 66005).

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, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By December 12, 2002, 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,[1] which is available at the Commission's PDR, located at One White Flint North, 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

[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 CFR 2.714(d), please see 67 FR 20884; April 29, 2002.

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, 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, 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, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: September 30, 2002.

Description of amendment request: The proposed amendment would revise the technical specification (TS) definition of containment integrity to ensure that all power-operated valves, relief valves, and check valves are included. The proposed changes would provide operability requirements to include the Type III containment isolation valves (CIVs), those valves that are in line with a containment isolation barrier consisting of a closed system within containment (*e.g.*, main steam isolation valves (MSIVs)). The proposed amendment would revise the applicability of CIV operability requirements for those plant conditions when containment integrity applies and the reactor is not critical. The proposed amendment would clarify that the exceptions to containment integrity provided in TS 3.6.1 apply equally to TS 3.6.2, whenever containment integrity is required. The proposed amendment would incorporate provisions for intermittent manual operation of the CIVs under administrative controls. The proposed amendment would also delete TS 4.8, ''Main Steam Isolation Valves,'' along with the reference to TS 4.8 in Table 4.1–2, Item No. 6. This change would delete a monthly requirement for a partial stroke test, but would not affect testing performed in accordance with the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which the licensee states would continue to ensure operability of the MSIVs. The proposed changes would also revise Figure 5–1, ''Extended Plot Plan,'' to correct inaccurate information, and Figure 5–3, ''Gaseous Effluent Release Points and Liquid Effluent Outfall Locations,'' and its accompanying table to reflect the modification which permanently isolated the liquid outfall associated with emergency discharge from Three Mile Island Nuclear Station, Unit 2.

Additional administrative and clerical changes are also included in the proposed TSs to delete obsolete references to TS sections that have been deleted, improve the consistency and clarity of the TSs, and revise the Bases of TS 3.1.6 to delete the setpoint range for emergency core cooling system cubicle leak detection and replace it with a single value.

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

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

Changes to the definition of containment integrity and the additional operability requirements for Containment Isolation Valves (CIVs) provide additional requirements and add clarity to the Technical Specifications. The addition of a provision for permitting intermittent opening of normally closed CIVs or manual control of power-operated CIVs under administrative control is consistent with the Standard Technical Specifications or a similar provision in the current TMI Unit 1 Technical Specifications. This assures that the containment will be isolated if necessary in the event of an accident previously evaluated and offsite dose from an accident will not be significantly increased. The additional operability requirements provide additional conservatism to the technical specifications.

None of the changes included with this License Amendment Request will result in any change to the configuration of plant components, affect any accident initiators associated with any accident previously evaluated or result in a significant increase in the offsite dose consequences of accidents previously evaluated. The administrative changes are needed to correct errors and the editorial changes will improve the clarity, consistency and readability of the Technical Specifications and do not affect the intent or interpretation.

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

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

The changes associated with this proposed amendment do not result in any additional hardware or design changes to structures, systems, or components (SCCs) of the plant; nor will any of these changes affect the ability of an SSC to perform its design function. No new failure mechanisms, malfunctions, or accident initiators will be introduced that were not considered in the design and licensing basis.

Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Additional operability requirements provide conservative improvements to the Technical Specifications. The addition of a provision for permitting intermittent opening of normally closed CIVs or manual control of power-operated CIVs under administrative control is consistent with the Standard Technical Specifications or with similar provisions in the current TMI Unit 1 Technical Specifications. This condition assures that the containment will be isolated if necessary in the event of an accident. Changes to the MSIV [main steam isolation valve] test requirements do not alter the Inservice Test requirements in accord[ance with] the American Society of Mechanical Engineers (ASME) [Boiler and Pressure Vessel] Code, which will continue to assure operability. The administrative changes are needed to correct errors and the editorial changes will improve the clarity, consistency, and readability of the Technical Specifications and do not affect the intent or interpretation.

None of the changes included with this request have the potential to significantly reduce a margin of safety. These changes do not affect the design of a plant component or instrument setpoint so as to [a]ffect its design basis or affect the controlling numerical value for any parameter established in the updated final safety analysis report or the license.

Therefore, operation of the facility in accordance with the 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: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: September 30, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 6.8.5, ''Reactor Building Leakage Rate Testing Program,'' to allow a one-time deferral of the next Type A, Containment Integrated Leak Rate Test (ILRT) from October 2003 to no later than September 2008.

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

The proposed revision to Technical Specification Section 6.8.5 (''Reactor Building Leakage Rate Testing Program'') involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test (1993). The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing guidelines invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications and NEI [Nuclear Energy Institute] 94–01. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. TMI, Unit 1 ILRT test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed revision to [the] Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing guidelines invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed revision to [the] Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific guidelines and conditions of the Reactor Building Leakage Rate Testing Program, as defined in [the] Technical Specifications, exist to ensure that the degree of reactor building containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by [the] Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications and NEI–94–01.

NUREG–1493 concludes that reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. Additionally, while Type B and C tests identify the vast majority (greater than 85%) of all potential leak paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing guidelines, the overall effect is very small. The TMI, Unit 1 plant specific risk analysis supports this conclusion. Therefore, the proposed Technical Specification 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: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: September 16, 2002.

Description of amendment request: The proposed change revises a license condition, contained in Appendix B of the Technical Specifications, to reflect a modification to support the implementation of an alternative source term (AST) on Unit 2 that would ensure seismic ruggedness of the alternate leakage treatment (ALT) piping and appendages. As a result of further modification development, it has been determined that only one check valve will be installed (*i.e.*, MVD–V5009) by the Unit 2 ALT piping modification. The proposed license amendment revises the affected license condition to require that only MVD–V5009 must be added to the facility check valve program.

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

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

Response: No.

The proposed change revises a license condition, added to Appendix B, ''Additional Conditions,'' of the Unit 2 Technical Specifications (TSs) in Amendment 246, which approved the implementation of Alternative Source Term. This license condition currently requires that alternate leakage treatment (ALT) path check valves MVD–V5008 and MDV–V5009 be included in the facility check valve program. Differences between the Unit 1 and Unit 2 main steam line isolation valve drain piping, which will be within the ALT pathway pressure boundary after a loss-of-coolant-accident (LOCA), obviate the need to install check valve MVD–V5008. This is because the Unit 2 steam bypass system was designed for full bypass capability and thus has two steam bypass chests; whereas Unit 1 has only one steam bypass chest. The Unit 2 design includes a drain line from the steam bypass chest, which ties into the same line that on Unit 1 was isolated post-LOCA by use of the 1–MVD–V5008 valve. Since, for Unit 2, the entire line is required to be seismically verified, up to and including the steam bypass chest, there was no benefit in installing the new check valve MVD–V5008 on Unit 2.

CP&L has performed an evaluation of the Unit 2 ALT path modification, in accordance with the provisions of 10 CFR 50.59, and determined that the modification can be implemented without prior NRC approval. As such, the requested amendment merely aligns the wording of the current license condition with the design of the Unit 2 ALT path modification. The original intent of the license condition was to ensure that check valves being installed as a result of the modification would be included in the facility check valve program. This intent is maintained by the proposed license condition. 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.

As stated above, CP&L has performed an evaluation of the Unit 2 ALT path modification, in accordance with the provisions of 10 CFR 50.59, and determined that the modification can be implemented without prior NRC approval. The requested amendment merely aligns the wording of the current license condition with the design of the Unit 2 ALT path modification. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change revises a license condition, added to Appendix B, Unit 2 TSs in Amendment 246. Therefore, the proposed change does not involve a significant reduction in a margin of safety. This license condition currently requires that ALT path check valves MVD–V5008 and MDV–V5009 be include in the facility check valve program. The proposed revision to affected Unit 2 license condition eliminates reference to a CP&L September 27, 2001, submittal and the requirement to include MVD–V5008 in the facility check valve program. The requested amendment merely aligns the wording of the current license condition with the design of the Unit 2 ALT path modification which has been evaluated, in accordance with the provisions of 10 CFR 50.59, and it has been determined that the modification can be implemented without prior NRC approval. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: September 26, 2002.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3.3.3.1, ''Monitoring Instrumentation, Radiation Monitoring,'' TS 3.3.4, ''Instrumentation, Containment Purge Valve Isolation Signal,'' TS 3.7.6.1, ''Plant Systems, Control Room Emergency Ventilation System,'' TS 3.9.4, ''Refueling Operations, Containment Penetrations,'' TS 3.9.8.1, ''Refueling Operations, Shutdown Cooling and Coolant Circulation—High Water Level,'' TS 3.9.8.2, ''Refueling Operations, Shutdown Cooling and Coolant Circulation—Low Water Level,'' and TS 3.9.15, ''Refueling Operations, Storage Pool Area Ventilation System.'' In addition, the TS Bases would be revised to address the proposed changes. The basis for the proposed changes is a re-analysis of the limiting design basis Fuel Handling Accident using an Alternative Source Term in accordance with Title 10 of the Code of Federal Regulations (10 CFR) section 50.67 and Regulatory Guide 1.183.

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.

The proposed changes involve the reanalysis of a Fuel Handling Accident (FHA) in the Containment, FHA in the Spent Fuel Pool Area, and the Cask Drop Accident in the Spent Fuel Pool Area. The new analyses, based on the Alternative Source Term (AST) in accordance with 10 CFR 50.67, will replace the existing analyses which are based on methodologies and assumptions derived from Regulatory Guide 1.25, Standard Review Plan (SRP) 15.7.4, SRP 15.7.5, and TID–14844. Because different methodologies are used, the new calculated doses are not directly comparable to the current calculated doses. If a consistent basis is used, it is expected that the new analyses assumptions in some cases result in a decrease in dose at the site boundary or to control room personnel and in some cases result in an increase in dose at the site boundary or to control room personnel. However, in all cases the analyses results are within the 10 CFR 50.67 and Regulatory Guide 1.183 acceptance criteria.

As a result of the new analyses, changes to the Technical Specifications are proposed which take credit for the new analyses. The proposed changes to the Technical Specifications modify requirements regarding Containment closure and Spent Fuel Pool area ventilation during movement of irradiated fuel assemblies in Containment and in the Spent Fuel Pool area. The proposed changes will allow Containment penetrations, including the equipment door and personnel airlock door, to be maintained open under administrative control. The proposed changes will eliminate the requirements for automatic closure of Containment purge during Mode 6 fuel movement. The technical specifications associated with storage pool area ventilation will be deleted. These proposed changes do not involve physical modifications to plant equipment and do not change the operational methods or procedures used for the physical movement of irradiated fuel assemblies in Containment or in the Spent Fuel Pool area. As such, the proposed changes have no effect on the probability of the occurrence of any accident previously evaluated.

The revised requirements apply only when irradiated fuel assemblies are being moved in Containment or the Spent Fuel Pool area. Previously evaluated accidents with the plant in other conditions including Modes 1 through Mode 5 are not impacted. The AST methodology is used to evaluate a FHA that is postulated to occur during fuel movement activities in Containment and in the Spent Fuel Pool area. The AST analyses follow the guidance of NRC Regulatory Guide 1.183 and the acceptance criteria of 10 CFR 50.67. The analyses demonstrate that the dose consequences meet the regulatory acceptance criteria.

The FHA Analyses conservatively assume that the Containment building and the fuel storage building, including ventilation filtration systems for those buildings do not diminish or delay the assumed fission product release. The analysis does take credit for, and technical specifications enforce, the presence of 23 feet of water over the irradiated fuel while fuel movement activities are being performed. The analysis also takes credit for, and the technical specification bases enforce a fuel decay time of at least 72 hours. In addition, administrative controls are put in place to provide for closure of Containment atmosphere boundary openings in the event of a FHA. Use of an alternative analysis method does not affect fuel parameters or the equipment used to handle the fuel. The above proposed changes to the Technical Specifications reflect assumptions made in the FHA Analyses. The other changes to the Technical Specifications are also consistent with the revised FHA Analyses. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment involves the use of an alternative analysis methodology for the evaluation of the dose consequences from a FHA that is postulated to occur in either the Containment or the Spent Fuel Pool area. The analysis demonstrates that Containment closure conditions and automatic closure of the Containment purge are not required to maintain dose consequence within regulatory limits following a postulated FHA inside Containment. Therefore, the new analysis supports proposed changes to requirements for Containment closure during movement of irradiated fuel assemblies in Containment. The analysis results also demonstrate that operation of the Spent Fuel Pool area ventilation system is not required to maintain dose consequences within regulatory limits following a postulated FHA in the Spent Fuel Pool area. The Containment closure components (*e.g.*, equipment door, personnel airlock doors, and various Containment penetrations) and filtration systems are not accident initiators. The proposed changes do not involve the addition of new systems or components nor do they involve the modification of existing plant systems. The proposed changes do not affect the way in which a FHA is postulated to occur. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The existing dose analysis methodology and assumptions demonstrate that the dose consequences of a FHA are within regulatory limits for whole body and thyroid doses as established in 10 CFR 100. The alternative dose analysis methodology and assumptions also demonstrate that the dose consequences of a FHA are within regulatory limits. The limits applicable to the alternative analysis are established in 10 CFR 50.67 in conjunction with the TEDE (total effective dose equivalent) acceptance criteria directed in Regulatory Guide 1.183. The acceptance criteria for both dose analysis methods have been developed for the purpose of evaluating design basis accidents to demonstrate adequate protection of public health and safety. An acceptable margin of safety is inherent in both types of acceptance criteria. 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Andersen (Acting).

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

Date of amendment request: September 30, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications for the plant's reactor building integrity. The proposed amendment would (1) modify the surveillance requirement to be consistent with the design of the reactor building access openings, (2) modify the frequency of the surveillance requirement for visual inspections for the exposed interior and exterior surface of the reactor building, and (3) modify the administrative controls for the containment leakage rate testing program.

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:

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment does not involve a significant hazard.

Conformance of the proposed amendment to the standards for a determination of no significant hazards, as defined in 10CFR50.92, is shown in the following:

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

No. The proposed amendment to the Technical Specifications does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the [Updated Final Safety Analysis Report] UFSAR. Therefore, the proposed amendment does not result in the increase in the probability or consequences of an accident previously evaluated.

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

No. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the facility which should introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators, since the containment and reactor building function primarily as accident mitigators.

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

No. Implementation of this amendment would not involve a significant reduction in the 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, including the performance of the containment and reactor building. The ability of the containment and reactor building to perform their design function will not be impaired by the implementation of this amendment at McGuire Nuclear Station. Consequently, no safety margins will be impacted.

Conclusion

Based on the preceding analysis, it is concluded that the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10 CFR 50.92.

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: August 29, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for the plants direct-current (DC) system batteries. The Surveillance Requirements for the current TS for DC sources require a battery service test to be performed each 18 months. A note provides that, on a once per 60 month frequency, the service test requirement may be met by performing a modified performance test. The TS change would remove the once per 60 month restriction, thus allowing the requirement for a service test to be met by a modified performance test that bounds the conditions of the service test. The licensee states that the proposed change will allow the use of a consistent battery testing technique in order to provide consistent data for trending battery performance.

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

Operation of the facilities in accordance with this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The Class 1E DC [direct-current] power system is not an initiator to any accident sequence analyzed in the Updated Final Safety Analysis Report. The safety features of the batteries will continue to function as designed and in accordance with all applicable TS. The design and operation of the system is not being modified by this proposed amendment. This amendment only revise[s] the requirements for testing the batteries. Therefore, there will be no impact on any accident probabilities or consequences.

Second Standard

Operation of the facilities in accordance with this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of this proposed amendment. No changes are being made to any structure, system, or component which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators and does not impact any safety analysis.

Third Standard

Operation of the facilities in accordance with this amendment would not involve a significant reduction in a margin of safety. The change to the battery surveillance will ensure each station's batteries are maintained in a highly reliable manner. The batteries will continue to be tested every 18 months with the modified performance test enveloping the service test. The equipment powered by the batteries will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

Based on the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazard consideration.

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.

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

Date of amendment request: September 19, 2002.

Description of amendment request: The proposed amendment would extend the allowable outage time (AOT) for the emergency diesel generators (EDGs) from 72 hours to a maximum of 14 days.

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 TS [technical specification] change does not affect the design, operational characteristics, function or reliability of the EDGs. The EDGs are not the initiators of previously evaluated accidents. The EDGs are designed to mitigate the consequences of previously evaluated accidents including a loss of offsite power. Extending the AOT for a single EDG would not affect the previously evaluated accidents since the remaining EDG supporting the redundant Engineered Safety Features (ESF) systems and the AACDG [alternate alternating current diesel generator], which has the capability to support either train of ESF systems, would continue to be available to perform the accident mitigating functions.

The duration of a TS AOT is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. A risk-informed assessment was performed which concluded that the increase in plant risk is small and consistent with the guidance contained in Regulatory Guide 1.177.

The current TS requirements ensure that redundant systems relying on the remaining EDG are operable. In addition to these requirements, administrative controls will be established to provide assurance that the AOT extension is not applied during adverse weather conditions that could potentially affect offsite power availability. Administrative controls are also implemented to avoid or minimize risk significant plant configurations during the time when an EDG is removed from service.

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

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

Response: No.

The proposed change does not involved a change in the design, configuration, or method of operation of the plant that could create the possibility of a new or different kind of accident. The proposed change extends the AOT currently allowed by the TS to 14 days. It also provides for a reduction to 72 hours, not to exceed 14 days, should the AACDG become inoperable during the extended AOT.

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 ESF systems required to mitigate the consequences of postulated accidents consist of two independent trains. The ESF systems on either of the two trains provide for the minimum safety functions necessary to shut down the unit and maintain it in a safe shutdown condition. Each of the two trains can be powered from one of the offsite power sources of its associated EDG. In addition, the AACDG is available to provide power to either or both of the two trains. This design provides adequate defense in depth to ensure that diverse power sources are available to accomplish the required safety functions. Thus, with one EDG out of service, there are sufficient means to accomplish the safety functions and prevent the release of radioactive material in the event of an accident.

The proposed change does not affect any of the assumptions or inputs to the Final Safety Analyses Report and does not erode the decrease in severe accident risk achieved with the issuance of the Station Blackout (SBO) Rule, 10 CFR 50.63, ''Loss of All Alternating Current Power.''

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 19, 2002.

Description of amendment request: The proposed amendment would extend the allowed outage time (AOT) for a single inoperable low pressure safety injection (LPSI) train from 72 hours to 7 days. In addition, an AOT of 72 hours would be included for other conditions where the equivalent of a single emergency core cooling system (ECCS) subsystem flow is still available to both the LPSI and high pressure safety injection (HPSI) trains. Also, if 100% of ECCS flow is unavailable due to two inoperable HPSI or LPSI trains, an action statement would been added to restore at least one of each HPSI and LPSI train to operable status within one hour.

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 HPSI and LPSI trains are part of the ECCS subsystem. Inoperable HPSI or LPSI components are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated. Both the HPSI and LPSI systems are primarily designed to mitigate the consequences of a Loss of Coolant Accident (LOCA). These proposed changes do not affect any of the assumptions used in the deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the LPSI AOT extension, probabilistic safety analysis (PSA) methods were utilized. The results of the analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. The analyses are detailed in CE NPSD–995, Combustion Engineering Owners Group *Joint Applications Report for Low Pressure Safety Injection System AOT Extension.*

The proposed change allows a combination of equipment from redundant trains to be inoperable provided that at least the equivalent flow of a single HPSI and LPSI train of ECCS remains operable. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Allowing equipment from redundant trains to constitute a single operable train does not increase the probability that a failure leading to an analyzed event will occur. The ECCS components are passive until an actuation signal is generated. This change does not increase the failure probability of the ECCS components. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

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

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

The proposed change does not change the design or configuration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, and the setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures, which ensure the plant remains within analyzed limits, is being proposed and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change will only provide the plant some flexibility in maintaining the minimum equipment required to be Operable to perform the ECCS function while in this Condition. The change does not alter assumptions made in the safety analysis and licensing basis.

Therefore, the proposed change will 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?

The CE NPSD–995 and ANO–2 [Arkansas Nuclear One, Unit 2] PSA evaluations demonstrate that the changes are essentially risk neutral or risk beneficial. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed change. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating a transient event. The proposed change, which allows operation to continue for up to 72 hours with components inoperable in both ECCS subsystems, is acceptable based on the remaining ECCS components providing 100% of the required ECCS flow.

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: December 12, 2001, as supplemented on October 10, 2002.

Description of amendment request: The proposed amendment would change the Technical Specification Tables 3.2.A, 3.2.B, 4.2.A, and 4.2.B. The proposed changes affect various instrument trip level settings and decreases the calibration frequencies for a variety of instruments. The proposed changes also involve clarifications to the Reactor Water Cleanup system trip configuration and the titles of certain trip systems. In addition, the proposed changes would make certain editorial and administrative corrections. The proposed setpoint changes and calibration frequencies are based on the licensee's evaluation.

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

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

The methodology used to determine the proposed trip level settings and surveillance intervals ensure adequate performance of the affected instrumentation. In addition, the affected instruments are not initiators of any accident previously evaluated. Therefore, the proposed trip level setting and surveillance intervals will not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes to trip level settings and surveillance intervals were establish using methodologies subject to 10 CFR Appendix B Quality Assurance program and ensure existing radiological limits are met. Therefore, the proposed trip level settings and surveillance intervals will not involve a significant increase in the consequences of an accident previously evaluated.

Other changes are editorial or administrative in nature and can not significantly increase the probability or consequences of an accident previously evaluated.

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

No new or different [kind] of accidents or malfunctions than those previously analyzed in Pilgrim's UFSAR [Updated Final Safety Analysis Report] are introduced by this proposed change because there are no new failure modes introduced. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to trip level settings and surveillance intervals were established using approved methodologies subject to a 10 CFR, Appendix B, Quality Assurance program and existing radiological limits are met. These changes do not impact Pilgrim's configuration or operation.

Editorial and administrative type changes do not impact the operation or configuration of Pilgrim. For the above reasons the proposed change does not result in 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: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: James W. Andersen, Acting.

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

Date of amendment request: July 5, 2002.

Description of amendment request: The proposed amendment would relocate the ''Primary System Boundary—Shock Suppressors (Snubbers),'' Technical Specifications (TS) 3/4.6.I, from the TS to the Updated Final Safety Analysis Report (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 proposed change is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. Snubbers are not assumed to be an initiator of any analyzed event, nor are they assumed in the mitigation of consequences of accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated[.]

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

Response: No. The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the UFSAR. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Andersen, Acting.

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

Date of amendment request: August 16, 2002.

Description of amendment request: The proposed amendment would relocate certain Control Rod Block functions from Technical Specifications 3/4.2.C, ''Instrumentation that Initiates Rod Blocks,'' to the Updated Final Safety Analysis Report.

Basis for proposed no significant hazards consideration determination:

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

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

Response: No.

The proposed change is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. These control rod blocks are not assumed to be an initiator of any analyzed event, nor are they assumed in the mitigation of consequences of accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident form any accident previously evaluated.

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

Response: No.

The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the FSAR [Final Safety Analysis Report]. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Andersen, Acting.

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: October 4, 2002.

Description of amendment request: Change the Technical Specifications by extending the primary containment integrated leak rate testing (ILRT) interval on a one-time basis from 10 years to no longer than approximately 10.6 years.

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 operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specifications 6.7.C ''Primary Containment Leak Rate Testing Program'' involves a one-time extension to the current interval for Type A containment testing. The current test interval of 10 years would be extended on a one-time basis to no longer than approximately 10.6 years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of anaccident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leak rate tests. Type B and C containment leak rate tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG[–]1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. VY's [Vermont Yankee] ILRT test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME [American Society of Mechanical Engineers] Section XI, the Maintenance Rule and Licensing commitments related to containment coatings serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment Leak Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leak rate tests. The proposed surveillance interval extension is bounded by the 15 month extension currently authorized within NEI [Nuclear Energy Institute] 94–01. Type B and C containment leak rate tests will continue to be performed at the frequency currently required by plant Technical Specifications. VY's, as well as the industries experience, strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR [Boiling Water Reactor] containments allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification 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: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Andersen, Acting.

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: August 16, 2002.

Description of amendment request: The proposed amendments would modify the Unit 3 allowable value, and the Units 2 and 3 surveillance requirements for the reactor protection system scram discharge volume water level-high function.

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.

Dresden Nuclear Power Station (DNPS), Unit 3 plans to implement a design change that upgrades the Scram Discharge Volume Water Level—High instrumentation from existing float-type level switches to electronic analog trip units. Analog trip units are a proven technology that is more reliable than existing equipment. The proposed design is consistent with a generic design that has been previously reviewed and approved by the NRC. Analog trip units are used in various applications at DNPS, including the Reactor Protection System (RPS) Low Water Level Trip Function.

The proposed Technical Specifications (TS) changes add new Unit 3 Channel Check and trip unit calibration Surveillance Requirements (SRs) for the new analog trip units associated with the Scram Discharge Volume Water Level—High RPS Trip Function. These new Unit 3 SRs are not applicable to the existing instrumentation because the existing float-type level switches are non-indicating and do not employ trip units. In addition, the proposed TS changes add a new trip unit calibration SR for existing Unit 2 and 3 instrumentation that is composed of differential pressure type level transmitter switches.

TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these proposed changes will not involve an increase in the probability of an accident previously evaluated. Additionally, these proposed changes will not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. The planned Unit 3 instrument upgrade is a more reliable design than existing equipment. The proposed changes establish requirements that ensure components are operable when necessary for the prevention or mitigation of accidents or transients. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

In summary, the proposed changes do not involve a significant increase in the probability or 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 proposed changes support a planned instrument upgrade on Unit 3 by incorporating SRs required to ensure operability. There is no change being made to the parameters within which DNPS is operated. The proposed changes do not adversely impact the manner in which the Scram Discharge Volume Water Level—High RPS instrumentation will operate under normal and abnormal operating conditions. The proposed changes will not alter the function demands on credited equipment. No alteration in the procedures, which ensure DNPS remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. Therefore, these proposed 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 methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these proposed 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.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms and actions. The proposed changes support a planned instrumentation upgrade to enhance the reliability of RPS instrumentation. The proposed changes do not affect the probability of failure or availability of the affected instrumentation. The revised Allowable Value, addition of a Channel Check and trip unit calibration, and revision of other SRs for RPS Instrumentation Channel Check and trip unit calibration, and revision of other SRs for RPS Instrumentation Function 7 (Scram Discharge Volume Water Level—High) are conservative changes that align the SRs for proper determination of operability with that of similar instrumentation. Therefore, it is concluded that 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 and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: October 15, 2002.

Description of amendment request: The proposed amendment modifies the reactor coolant system flow rate from 363,000 gallons per minute (gpm) to 355,000 gpm in Saint Lucie Unit 2 Technical Specifications (TS) Table 3.3–2 and a footnote for Table 2.2–1.

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

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

The proposed amendment would decrease the value of design reactor coolant system flow rate. This reduction in the reactor coolant system (RCS) flow requirement will support operation of the plant with an increased steam generator (SG) tube plugging. The changes to the Technical Specification (TS) bases either support the proposed flow reduction or are administrative in nature, consistent with the current design basis. The parameters affected by the proposed changes are not accident initiators and do not affect the frequency of occurrence of previously analyzed transients. Additionally, there are no changes to any active plant component.

This evaluation has demonstrated acceptable results for all the accidents previously analyzed. It is concluded that the radiological consequences would remain within their established acceptance criteria when including effects of the proposed reduction in the RCS flow, which would support an increased steam generator tube plugging level.

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

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

This proposed amendment revises the RCS design flow requirement to cover plant operation with increased steam generator tube plugging. There are no physical changes to the plant systems or system interactions due to the proposed changes. The modes of operation of the plant and the design functions of all the safety systems remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The impact of the proposed changes on the design basis accident analysis was evaluated and it is concluded that the setpoint and safety analyses of all design basis accidents meet the applicable acceptance criteria with respect to the radiological consequences, specified acceptable fuel design limits (SAFDL), primary and secondary overpressurization, peak containment pressure and temperature, and 10 CFR 50.46 requirements.

Therefore, operation of the facility in accordance with the proposed amendment would 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Allen G. Howe.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: October 21, 2002.

Description of amendment request: The proposed amendment deletes the requirements defined in Technical Specification (TS) 3/4.9.3, ''Refueling Operations, Decay Time,'' and places them in the TS Bases. Additionally this amendment proposes to modify the TS Bases definition of ''recently irradiated fuel'' will be re-defined as fuel that has occupied part of a critical reactor core within the previous 72 hours.

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

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

No. The accident of concern related to the proposed change is the fuel handling accident (FHA). This accident assumes a dropped fuel assembly. One of the assumptions made in the analysis is that fuel movement is delayed for some time period after shutdown to accommodate cooldown of the reactor coolant system and disassembly of the reactor pressure vessel. This delay period allows for radioactive decay of the in-reactor vessel fission product inventory. Reducing the analyzed decay time from 100 hours to 72 hours does not increase the probability of a FHA because the timing of fuel movement in the reactor pressure vessel does not alter the manner in which fuel assemblies are handled.

Reducing the analyzed decay time from 100 hours to 72 hours does increase the offsite dose and control room dose projections of a FHA above those previously reviewed and approved by the NRC for Turkey Point Units 3 and 4 per Amendments 216 and 210. However, it has been shown by reanalysis of such an accident involving irradiated fuel with at least 72 hours of decay that the projected doses remain well within applicable regulatory limits. Hence, the proposed change in timing of fuel movement in the reactor pressure vessel does not involve a significant increase in the consequences of a FHA.

Additionally, the manner in which the minimum in-reactor vessel decay time is controlled will not impact the probability of occurrence, or the consequences of a FHA. Relocating the decay time requirement from the TS to the TS Bases document and other administrative controls will continue to ensure that this key accident analysis assumption is upheld. The inherent delay associated with completing the required preparatory steps for moving fuel in the reactor vessel further ensures that the proposed 72-hour decay time will be met for a refueling outage.

Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The impact of the proposed change is limited to fuel handling operations and spent fuel pool cooling. No physical plant changes are proposed to accommodate the timing change for fuel movement. Hence, no new failure modes are created that would cause a new or different kind of accident from any accident previously evaluated. The supporting analysis for the timing change demonstrates that the associated increase in decay heat load will not cause any spent fuel pool (SFP) component or structure to operate outside design limits. Adequate margins to safety are maintained with respect to SFP water temperature and structural loading.

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.

Additionally, the manner which the minimum in-reactor vessel decay time is controlled will not impact the operation of any structure, system, or component.

Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed change in plant operation does not significantly reduce the margin of safety. It has been shown by reanalysis of a FHA involving irradiated fuel with at least 72 hours of decay that the projected doses will be well within applicable regulatory limits. Additionally, it has been shown by thermal hydraulic analysis that operation of the SFP cooling system in accordance with the restrictions and limitations identified in the amendments application will maintain adequate margins to pool boiling. Analysis of transient SFP concrete temperatures similarly demonstrates that the integrity of the pool structure will not be compromised if the amount of in-reactor vessel fuel assembly decay time is reduced from 100 hours to 72 hours.

The proposed change in the manner in which the minimum in-reactor vessel decay time will be controlled will not impact plant safety. Relocating the decay time requirement from the TS to the TS Bases document and other administrative controls will continue to ensure that this key accident analysis assumption is upheld. The inherent delay associated with completing the required preparatory steps for moving fuel in the reactor vessel further ensures that the proposed 72-hour decay time will be met for a refueling outage.

Therefore, operation of the facility in accordance with the proposed amendments 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: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: October 16, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation Trip Setpoints.'' The proposed changes are part of a planned design change to replace the existing 4160 volt (4kV) offsite power transformers, loss-of-voltage relays, and degraded voltage relays with components of an improved design to increase the reliability of offsite power for safety-related equipment.

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated

The proposed changes to the degraded voltage and loss-of-voltage setpoints and time delay affect when an emergency bus that is experiencing low or degraded voltage will trip from offsite power and shift to an emergency diesel generator. While the setpoints that initiate this action will be modified, the function remains the same. The setpoints have been analyzed to ensure spurious trips will be avoided. The proposed changes will not significantly affect any accident initiators or precursors. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated

The proposed changes to the degraded voltage and loss-of-voltage setpoints and time delay affect when an emergency bus that is experiencing low or degraded voltage will trip from offsite power and shift to an emergency diesel generator. While the setpoints that initiate this action will be modified, they are bounded by the current safety analysis. The function of the plant equipment remains the same. The proposed changes improve the reliability of safety-related equipment to operate as designed. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the consequences of an accident previously analyzed are not significantly increased.

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

Response: No.

The proposed changes to the degraded voltage and loss-of-voltage setpoints and time delay do not affect existing or introduce any new accident precursors or modes of operation. The relays will continue to detect undervoltage conditions and transfer safety loads to the emergency diesel generators at a voltage level adequate to ensure proper safety equipment performance and to prevent equipment damage. The function of the relays remains the same. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed changes will allow all safety-related loads to have sufficient voltage to perform their intended safety function while ensuring spurious trips are avoided. Thus, the results of the accident analyses will not be affected as the input assumptions are protected. The format changes are intended to improve readability, consistency with NUREG–1431, Revision 2, and appearance. In addition, they do not alter any requirements. The bases change provides explanatory information only. Thus, the proposed changes do not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, [Indiana Michigan Power Company] I&M has concluded that the proposed changes involve 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 amendment requests involve no significant hazards consideration.

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: October 7, 2002.

Description of amendment request: The proposed amendment would add Specification 4.0.3 to address missed surveillances. This new specification specifies an initial 24-hour delay period for performing a missed surveillance prescribed by Specification 3.0.3. Specification 4.0.3 will also require: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' In addition, the licensee proposed to add wording to each of the following existing specifications such that the new Specification 4.0.3 would apply to them: Specification 6.16, 6.17, 6.18, and 6.19.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714).

The licensee affirmed the applicability of the following NSHC determination in its application dated October 7, 2002. The NSHC determination is restated below.

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

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in [a] margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on [a] margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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

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

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The proposed amendment will change the Limiting Condition for Operation (LCO) 2.3(2).a, ''Emergency Core Cooling Systems,'' for the allowed outage time (AOT) for a single train of the low pressure safety injection system. The proposed change is based on the Combustion Engineering Owners Group Topical Report CE NPSD–995, ''Joint Applications Report for Low Pressure Safety Injection System AOT Extension.'' This amendment will permit the licensee to extend the AOT for a single low pressure safety injection (LPSI) train from the existing 24 hours to 7 days.

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 allowed outage time is not an initiator of any previously evaluated accident. The proposed change to the allowed outage time for a single LPSI train will not prevent the safety systems from performing their accident mitigation function as assumed in the safety analysis.

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

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

The proposed change only affects the technical specifications and does not involve a physical change to the plant. Modifications will not be made to existing components nor will any new or different types of equipment be installed. The proposed change modifies the allowed outage time for a single LPSI train from 24 hours to 7 days for the purpose of performing preventive or corrective maintenance, or surveillance testing. Actions will be taken to ensure the increase in LPSI allowed outage time is incorporated appropriately into plant procedures.

Therefore, this 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 modifies the allowed outage time for a single LPSI train to permit necessary ECCS [emergency core cooling system] maintenance or testing to be performed in a measured, deliberate fashion. Results of an integrated assessment of the overall plant risk associated with the adoption of the proposed AOT extension show a negligible increase in plant risk. The increase in allowed outage time will also permit more efficient and more safely managed plant operations and can help reduce the risk associated with changing plant operating modes.

An evaluation of the impact of extending the AOT for a single LPSI train on plant risk was performed for the conditions of the plant being at power. While at power, the incremental conditional core damage frequency (ICCDF) was determined to be 1.396E–05 per year, with a 5.80E–07 per year incremental increase in the core damage frequency attributed to extending the allowed outage time from 24 hours to seven days.

A sensitivity analysis was performed to identify the impact on core damage probability over a seven day interval that results from performing maintenance on one LPSI train while in a shutdown mode. Results of this study show that even small improvements in LPSI train reliability will produce a decrease in core damage probability, thus the net impact of performing LPSI train preventive maintenance while at power is risk-beneficial.

The unavailability of one LPSI train resulted in a large early release frequency of 2.636E–06 per year, with a 2.40E–08 per year incremental conditional large early release frequency (ICLERF) attributed to extending the allowed outage time from 24 hours to seven days.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The proposed amendment would relocate the requirements of Technical Specification (TS) 2.13, ''Nuclear Detector Cooling System,'' to the Fort Calhoun Station Updated Safety Analysis Report (USAR). The accident analyses do not assume operation of the nuclear detector cooling system; therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. The requirements will be relocated to the USAR.

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 relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in the TS set forth in 10 CFR 50.36(c)(2)(ii). The requirements for Nuclear Detector Cooling are being relocated from TS to the USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the 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.

The proposed change relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements for Nuclear Detector Cooling remain the same as the existing TS. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The proposed amendment would increase the amount of diesel fuel oil required by Technical Specification (TS) 2.7, ''Electrical Systems,'' to be kept in the auxiliary boiler fuel oil storage tank. A recent calculation determined that the amount of diesel fuel oil required by TS 2.7 is slightly insufficient (35 gallon shortfall) for 7 days of emergency diesel generator (EDG) operation.

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

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

No changes to the EDG diesel fuel oil storage and distribution system configuration or usage is required to achieve the inventory increase. This change only increases the current minimum inventory requirements listed in TS 2.7 and assures that the inventory will meet the capacity requirements of IEEE–308, which requires sufficient fuel for 7 days of EDG operation following the most severe accident. Increasing the minimum inventory requirement of FO–10, the auxiliary boiler fuel oil tank by 2000 gallons enables the site to meet this criterion and provides an extra margin of inventory to prevent any future concerns.

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

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

No changes to the Emergency Diesel Generator fuel oil storage and distribution system configuration or usage are required to achieve the inventory increase. FO–10 has a capacity of 18,000 gallons. Therefore, FO–10 can readily accommodate the additional 2000 gallons of inventory. Therefore, the proposed changes will 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 will increase the margin of safety by requiring that additional diesel fuel oil inventory be kept on-site to ensure that the 7 day on-site fuel supply criteria is met.

Therefore, this technical specification change does not involve a reduction in the margins 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The proposed amendment will relocate the requirements of Technical Specification (TS) 3.5(5), ''Surveillance for Prestressing System,'' for testing prestressed concrete containment tendons to the Fort Calhoun Station (FCS) Updated Safety Analysis Report (USAR). This proposed amendment will also add a TS requirement (TS 5.21) for a containment tendon testing program consistent with that presented in Section 5.5 of NUREG–1432, ''Improved Standard Technical Specification (ITS) for Combustion Engineering Plants.''

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 relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in the TS set forth in 10 CFR 50.36(c)(2)(ii). The requirements for testing Prestressed Concrete Containment Tendons are being relocated from TS to the USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the 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.

The proposed change relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements for testing Prestressed Concrete Containment Tendons remain the same as the existing TS. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The proposed amendment will change Technical Specification 5.19, ''Containment Leakage Rate Testing Program,'' to extend the integrated leak rate test (ILRT) surveillance interval from 10 to 15 years. The proposed changes are justified based on a combination of risk-informed analysis and assessment of the containment structural condition utilizing ILRT historical results and containment inspection programs.

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

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

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change adds a one-time extension to the current surveillance interval for Type A testing (ILRT). The current test interval of 10 years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A test is not a modification, nor a change in the way that plant systems, structures, or components are operated, and is not an activity that could lead to equipment failure or accident initiation. The proposed change does not involve a significant increase in the consequences of an accident since research in Reference 10.3 [NUREG–1493, Performance Based Containment Leak-Test Program] has found that generically very few potential leaks are not identified in Type B and C tests. Reference 10.3 concluded that an increase in the test interval to 20 years resulted in an imperceptible increase in risk. FCS provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner only detectable by Type A testing. Inspections required by ASME code and the Maintenance Rule are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by 10 CFR 50, Appendix J are not affected by this proposed extension to the Type A test interval and will continue to identify containment penetration leakage paths that would otherwise require a Type A test.

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

The proposed change adds a one-time extension to the current surveillance interval [* * *] for Type A testing (ILRT). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the 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.

The proposed change will not result in operation of the facility involving a significant reduction in a margin of safety. The proposed change adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. Reference 10.3 has found that generically very few potential leaks are not identified in Type B and C tests. Reference 10.3 concluded that an increase in the test interval to 20 years resulted in an imperceptible increase in risk. Furthermore, the extended test interval would have a minimal effect on such risk since Type B and C testing detect over 95 percent of potential leakage paths. A plant specific risk calculation, as part of Reference 10.2, [WCAP–15691, Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension, Revision 3, August 2002] on this topic obtained results consistent with the generic conclusions of Reference 10.3. The overall increase in risk contribution was determined as 0.31%.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 24, 2002.

Description of amendment request: This proposed license amendment request would revise Technical Specification (TS) 4.8.1.1, ''AC Sources'' and the associated Bases section related to the Emergency Diesel Generators (EDG). This change would clarify the requirement for the start time test performed on a 184 day and an 18-month frequency. The proposed change will revise Surveillance Requirement (SR) 4.8.1.1.2.f.1 and 4.8.1.1.2.g.5 to more accurately reflect the plant conditions during EDG start testing.

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 change does not involve a significant increase in probability or consequences of an accident previously evaluated?

This proposed amendment modifies an EDG Surveillance Requirement and does not impact the offsite AC distribution system; therefore the probability of any LOOP [loss of off-site power], including one concurrent with a LOCA [loss-of-coolant accident] is not significantly increased.

The proposed change revises the SR to better match the plant conditions during the test. SR 4.8.1.1.2.f.1 and 4.8.1.1.2.g.5 are performed with the EDG unloaded and as a result, overshoots its target nominal voltage and frequency during the test. In an actual event, the EDG would be almost immediately loaded once minimum voltage and frequency requirements are satisfied, thereby minimizing the overshoot.

To ensure the EDGs are capable of fulfilling their safety function, the proposed SR requires EDG voltage and frequency to achieve the specified minimum acceptable valued within 10 seconds, and to settle to a steady state voltage and frequency within the minimum and maximum values. That is, the upper limits are only applicable for steady state operation and do not apply during the transient portion of the EDG start. This change revises the acceptance criteria of 4.8.1.1.2.f.1 and 4.8.1.1.2.g.5 to clarify which voltage and frequency limits are applicable during the transient and steady state portions of the EDG start test.

This change does not affect the EDGs ability to supply the minimum voltage and frequency within 10 seconds or the steady state voltage and frequency required by the FSAR [Final Safety Analysis Report]. The EDGs will continue to perform their intended safety function, in accordance with the safety analysis. Thus, the consequences of any previously analyzed event are not significantly increased by this change.

The proposed change to 3.8.1.1, Action b.2 will not increase the probability or consequences of an accident previously evaluated. The change to this requirement to allow determination of no common cause failure mechanism has no impact on any accident. This change allows for not testing the redundant EDG if it can be demonstrated the failure mechanism of the affected EDG is not common cause. The normal TS surveillance testing schedule assures that operable EDG(s) are capable of performing their intended safety functions. The revision to the footnote on page 3/4.8–1 assures the action will be completed even if the EDG is restored to operable status within the action completion time.

The proposed revision to the fuel oil surveillance program will not preclude the EDGs from fulfilling their design functions. These changes provide flexibility to the testing program and continue to provide assurances that the fuel oil is acceptable for sustained engine operation. Eliminating or revising methodologies for testing of the fuel oil will not increase any probabilities or consequences to any 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 change revises SR 4.8.1.1.2.f.1 requirements to clarify which voltage and frequency limits are applicable during the transient and steady state portions of the EDG start testing. No changes are being made in equipment hardware or software, operational philosophy, testing frequency, how the system actually operates, or how the system is physically tested. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The elimination of unnecessary surveillance testing does not affect the design bases of the EDGs. The EDGs are designed to provide electrical power to the equipment important for safety during all modes and plant conditions following a loss of offsite power. The proposed changes to the Action requirements are consistent with NUREG–1431, NUREG–1366, Generic Letter 93–05, industry operating experience, and VCS operating experience. These changes are intended to improve plant safety, decrease equipment degradation, and remove unnecessary burden on personnel resources by reducing the amount of testing that the TS requires during power operation.

The revision to the fuel oil testing methodology does not impact the capabilities or functions of the EDGs. This testing methodology change will continue to assure the EDG is not degraded due to the fuel oil used. Existing test methodologies and guidance will continue to be followed, unless an evaluation demonstrates another methodology is as effective. Since the changes do not adversely impact important to safety equipment that is used in mitigating an accident, they will not create the possibility of an accident different from any previously evaluated.

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

The EDGs will still perform their intended safety function, in accordance with the VCSNS accident analysis. The revised test acceptance criteria are a much better match for the tested condition (unloaded). The performance of other TS SRs (in particular 4.8.1.1.2 g.4.b, 4.8.1.1.2g.6 and 4.8.1.1.2g.14) demonstrate EDG operability in conditions that are more representative of postulated accident conditions (loaded in the actual time sequence assumed in the accident analysis). The proposed amendment does not alter any acceptance criteria or equipment testing scope, which could impact the accident analysis.

The proposed change to exempt specific surveillance testing, as long as potential common cause can be ruled out, and eliminate unnecessary mechanical stress and wear on the diesel generator is an effort to improve plant reliability and safety. These changes are consistent with NUREG–1431, NUREG–1366, industry operating experience, and VCS operating experience and do not adversely affect the design bases, accident analysis, reliability or capability of the EDGs to perform their intended safety function. The revised footnote will assure that once the action is initiated, it will be completed regardless of when the EDG is restored to operability.

The proposed change to the fuel oil testing methodology has no impact on any safety margin. Accident analysis requires that the EDGs provide electric power to selected components during an accident scenario. The fuel oil quality will continue to meet established acceptance criteria and support the design function of the EDGs.

Since the design and licensing basis of the plant is unaffected, the 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: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski.

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 amendment request: April 4, 2002.

Description of amendment request: The proposed amendments would 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.

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

The proposed revision to Technical Specifications 5.5.17, ''Containment Leakage Rate Testing Program,'' involves a one time extension to the current interval for Type A containment leak testing. The current test interval of ten (10) years would be extended on a one time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. FNP [Joseph M. Farley Nuclear Plant] test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanism which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME [American Society of Mechanical Engineers] Section XI, the Maintenance Rule and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specifications change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed revision to Technical Specifications involves a one time extension to the current interval for Type A containment leak testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specifications change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specifications change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed revision to Technical Specifications involves a one time extension to the current interval for Type A containment leak testing. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rates limits specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

FNP and industry experience strongly support the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specifications 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

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 amendment request: September 24, 2002.

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) Limiting Conditions for Operation 3.7.10, Control Room Emergency Filtration/Pressurization System; and associated Bases. These changes will allow maintenance on ventilation area pressure boundaries (*i.e.*, doors) that cannot be conducted within the requirements of existing TS. The changes are based on U.S. Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force—287, Rev. 5. In addition, the proposed amendments would revise TS 3.7.12 to eliminate a requirement to cease power operation if the fuel handling accident function of the penetration room filtration system is inoperable.

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

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

The control room emergency filtration/pressurization system (CREFS) and the penetration room filtration (PRF) system are not initiators of any accident. The proposed changes do not alter the physical plant nor do they alter modes of plant operation. Therefore, the proposed changes do not affect the probability of any accident previously evaluated. Compensatory actions such as the availability of self-contained breathing apparatus or iodine filters provide additional assurance that the requirements of GDC [General Design Criteria] 19 are met. Prohibiting movement of irradiated fuel, or loads over irradiated fuel or core alterations when the control room boundary is inoperable and limiting movement of irradiated fuel or loads over the fuel in the spent fuel pool room when its boundary is inoperable will eliminate the potential for exceeding GDC 19 due to a fuel handling accident. These actions will also prevent an off site dose release in excess of analyzed values. 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 previously evaluated?

The CREFS and the PRF systems are not initiators of any analyzed accident. The proposed changes do not alter the operation of the plant or any of its equipment, introduce any permanent new equipment, adversely impact maintenance practices or result in any new failure mechanisms or single failures. Any temporary equipment utilized for compensatory measures will be subject to existing administrative controls that address issues such as fire prevention and seismic concerns. Therefore, there is no potential for a new accident and no potential for changing the progression of an analyzed accident. 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 result in a significant reduction in a margin of safety?

The proposed changes do not adversely affect the ability of the fission product barriers to perform their functions. Adequate compensatory measures are available to mitigate a breach in the control room, spent fuel pool room and penetration room pressure boundaries. The probability of a loss of coolant accident that would place demands on these systems during a period that the ventilation system pressure boundaries would be allowed to be inoperable has been shown to be very small. In addition, proposed administrative controls eliminate the potential for a fuel handling accident, with potential to exceed dose limits, while the spent fuel pool room boundary room is breached. Therefore, the proposed changes do not result in 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: September 3, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) 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; (2) modifying the current TS 6.8 by adding a new subsection 6.8.j, which will include the NUREG–1431, Revision 2, ISTS wording for TS 5.5.14 that discusses the TS Bases Control Program; and (3) modifying the ISTS wording, adopted in item 1 above, to allow a delay period of 24 hours or up to the surveillance frequency interval, whichever is greater, and to require a risk analysis to be performed for any surveillance greater than 24 hours.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the Consolidated Line Item Improvement Process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). Tennessee Valley Authority reviewed the following proposed NSHC determination published in the **Federal Register** as part of the CLIIP for Technical Specification Task Force (TSTF)–358, and concluded in its application of September 3, 2002, that the proposed NSHC determination applied to Sequoyah.

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:

Adoption of TSTF–358, Revision 6—Missed Surveillances

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does 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 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. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

In addition to the above determination of NSHC, the licensee has provided its analysis for the following proposed NSHC determination:

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 for the adoption of NUREG–1431, Revision 2, for Surveillance Requirements 3.0.1 and 3.0.3 wording and for the adoption of NUREG–1431, Revision 2, Technical Specification Bases Control Program, both of which are presented below:

Adoption of NUREG–1431, Revision 2, for Surveillance Requirements 3.0.1 and 3.0.3 Wording

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

The proposed change involves rewording of existing Specification 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2. These modifications involve no technical changes to the existing TS [Technical Specifications]. This change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does 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 Previously Evaluated

The proposed change involves the rewording of the existing Specification 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2. The change does not involve a physical alteration of the plant (no new or different type of equipment installed) or changes in the methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. Therefore, the proposed change does not create the probability of a new or different kind of accident from any accident previously evaluated.

C. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change involves rewording of the existing Specification 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2. The change is administrative in nature and will not involve any technical changes. The change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Since this change is administrative in nature, no question of safety is involved. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Adoption of NUREG–1431, Revision 2, Technical Specification Bases Control Program

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

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into the SQN [Sequoyah Nuclear Plant] Units 1 and 2 TS. This change involves no technical change to existing TS, it simply adds wording on how the bases section of the TS will be maintained and controlled. This change is administrative in nature and does not affect initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences or an accident previously evaluated.

B. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into the SQN Units 1 and 2 TS. The change does not involve a physical alteration of the plant (no new or different type of equipment installed) or changes in the methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. Therefore, the proposed change does not create the probability of a new or different kind of accident from any accident previously evaluated.

C. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into SQN Units 1 and 2 TS. The change is administrative in nature and will not involve any technical changes. The change will not reduce a margin of safety because they have not impact on any safety analysis assumptions. Since this change is administrative in nature, no question of safety is involved. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

The NRC staff 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.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: October 1, 2002.

Description of amendment request: The amendment would add a phrase to Limiting Condition for Operation (LCO) 3.1.8, ''Physics Tests Exceptions—Mode 2,'' of the technical specifications (TSs). The phrase to be added is that the number of required channels for certain functions in Table 3.3.1–1 of LCO 3.3.1, ''RTS Instrumentation,'' may be reduced from four to three required channels. LCO 3.1.8 applies to reactor Mode 2 during physics tests.

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 [for the proposed change] will remain within the bounds of the previously performed accident analyses since there are no permanent hardware changes. The design of the RTS [reactor trip system] instrumentation will be unaffected; only the manner in which the system is connected for short duration physics testing is being changed to allow the temporary bypass of one power range channel. The reactor protection system will continue to function in a manner consistent with the plant design basis since a sufficient number of power range channels will remain OPERABLE to assure the capability of protective functions, even with a postulated single failure. [The number of required channels for certain functions in Table 3.3.1–1 is only being reduced from 4 to 3 channels.] All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change will allow the temporary bypass of one power range neutron flux channel during the performance of low power physics testing in MODE 2. This results in a temporary change to the coincidence logic from one-out-of-three under the current TS (with a trip imposed on the channel used for physics testing) to two-out-of-three under the proposed TS (the channel used for physics testing would be in a bypassed state). However, this two-out-of-three coincidence logic still supports [the] required protection and control system applications, while reducing plant susceptibility to a spurious reactor trip.

The proposed change will not affect the probability of any event initiators. 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 [Wolf Creek Updated Safety Analysis Report].

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 permanent 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 power operation or change any operating parameters. No performance requirements will be affected.

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

The proposed amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System (other than as discussed above), or Solid State Protection System used in the plant protection systems. [The number of the required channels is not an initiator of an accident.]

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.

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 protective 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 (F'H), 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.

The proposed change does not eliminate any RTS surveillance or alter the Frequency of surveillances required by the Technical Specifications. The nominal RTS and Engineered Safety Features Actuation System (ESFAS) trip setpoints (TS Bases Tables B 3.3.1–1 and B 3.3.2–1), RTS and ESFAS allowable values (TS Tables 3.3.1–1 and 3.3.2–1), and the safety analysis limits assumed in the transient and accident analyses [(USAR Table 15.0–4)] are unchanged. None of the acceptance criteria for any accident analysis is changed. The potential reduction in the frequency of spurious reactor trips would effectively increase the margin of safety or, at a minimum, be risk-neutral.

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

NRC Section Chief: Stephen Dembek.

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, 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, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: April 26, 2002, as supplemented on July 11 and September 12, 2002.

Brief description of amendment: The amendment revised Sections 2.3, ''Limiting Safety System Settings,'' 3.1, ''Protective Instrumentation,'' and 3.10, ''Core Limits,'' of the Technical Specifications, and approved the use of flow control reference cards to support implementation of the Boiling Water Reactor Owners Group Option II solution for the long-term reactor stability problem.

Date of Issuance: October 18, 2002.

Effective date: October 18, 2002, and shall be implemented within 30 days of issuance.

Amendment No.: 235.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

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

The July 11 and September 12, 2002, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 18, 2002.

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: July 19, 2002, as supplemented September 6, 2002.

Brief description of amendment: The amendment revises Technical Specification (TS) Surveillance Requirement (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'' 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.'' The amendment also makes administrative changes to SRs 4.0.1 and 4.0.3 to be consistent with NUREG–1432, Revision 2, ''Standard Technical Specifications, Combustion Engineering Plants.''

Date of issuance: October 15, 2002.

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

Amendment No.: 271.

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

*Date of initial notice in **Federal Register:*** August 22, 2002 (67 FR 54497).

The September 6, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 15, 2002.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: July 29, 2002.

Brief description of amendments: The amendments revised Technical

Specification Surveillance Requirement 3.7.2.2 to decrease the allowable closure time for the turbine stop valves from 15 seconds to 1 second.

Date of Issuance: October 24, 2002.

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

Amendment Nos.: 329, 329, 330.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56320).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 18, 2002.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.0.3 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 or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: October 8, 2002.

Effective date: October 8, 2002, to be implemented within 60 days from the date of issuance.

Amendment No.: 180.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56321).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 17, 2001, as supplemented by letters dated February 26, August 14 and September 13, 2002.

Brief description of amendments: The amendments revise (1) Technical Specification (TS) Section 1.1, ''Definitions,'' for Dose Equivalent I–131, to allow the use of the thyroid dose conversion factors, listed in the International Commission on Radiological Protection Publication 30, ''Limits for Intakes of Radionuclides by Workers,'' and (2) Section 3.9.4, ''Containment Penetrations,'' to allow the equipment hatch, personnel air lock doors, and emergency air lock doors to remain open during core alterations and movement of irradiated fuel assemblies.

Date of issuance: October 21, 2002.

Effective date: October 21, 2002, to be implemented within 30 days from the date of issuance, including the completion of the administrative procedures that ensure that closure of the open containment penetrations, with direct access to the outside atmosphere during refueling operations with core alterations or irradiated fuel movement inside containment, will be initiated immediately in the event of a fuel handling accident inside containment, or if severe weather warnings are in effect.

Amendment Nos.: Unit 1—155; Unit 2—155.

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

*Date of initial notice in **Federal Register:*** January 8, 2002 (67 FR 929).

The supplemental letters dated February 26, August 14 and September 13, 2002, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 2002.

No significant hazards consideration comments received: No.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: November 15, 2001 as supplemented by letters dated January 31, July 31, and October 3, 2002.

Brief description of amendment: The amendment revises License Condition 2.C(10), ''Loading of Fuel into Casks in the Fuel Building,'' to license number NPF–1 for the Trojan Nuclear Plant (TNP). Specifically, these design changes are the result of the licensee's selection of Holtec International's design components (*e.g.*, the Multi-Purpose Cannister versus the Pressurized Water Reactor Basket. The new design basis limits impact the cask loading operations and contingency unloading in the Fuel Building.

Date of issuance: October 21, 2002.

Effective date: As of the date of issuance to be implemented and shall be implemented prior to placing Holtec International MPC's in the TNP ISFSI.

Facility Operating License No. NPF–1: The amendment changes the cask loading and contingency unloading operations in the Fuel Building.

*Date of initial notice in **Federal Register:*** April 2, 2002 (67 FR 15626).

The January 31, July 31, and October 3, 2002, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** (67 FR 15626) notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 21, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 4th day of November 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

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