Week of February 24, 2003—Tentative

Monday, February 24, 2003

2 p.m. Meeting with National Association of Regulatory Utility Commissioners (NARUC) (public meeting)

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

Week of March 3, 2003—Tentative

Monday, March 3, 2003

10 a.m. Briefing on status of Office of Nuclear Material Safety and Safeguards (NMSS) programs—Waste Safety (public meeting) (contact: Claudia Seelig, 301–415–7243)

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

2 p.m. Discussion of security issues (closed—Ex. 1)

Week of March 10, 2003—Tentative

There are no meetings scheduled for the week of March 10, 2003.

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

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

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

Dated: January 30, 2003.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–2713 Filed 1–31–03; 2:18 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, January 10, 2003, through January 23, 2003. The last biweekly notice was published on January 21, 2003 (68 FR 2796).

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

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 6, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714,[1] which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the

[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.

designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois; Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois; Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York County, Pennsylvania; Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: December 20, 2002.

Description of amendment request: Nuclear Regulatory Commission (NRC) Regulatory Issue Summary 2002–05: ''NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program,'' provides guidance on implementing the boiling water reactor (BWR) reactor pressure vessel integrated surveillance program (ISP). The amendment will modify the Updated Safety Analysis Reports (USARs) by removing the current facility reactor material surveillance capsule removal schedules from the facility USARs and specifying that these facilities will participate in an ISP developed by the BWR Vessel and Internals Project (BWRVIP). In addition, the Limerick Station will remove the current facility reactor material specimen surveillance schedule from the Technical Specifications.

With the exception of Oyster Creek, the USARs of each of the listed facilities contain a withdrawal schedule for the reactor pressure vessel material specimens. For those facilities which are not scheduled to remove a material specimen as part of the ISP (*i.e.*, Clinton, Quad Cities, and Limerick), the proposed amendment would remove these plant-specific schedules from the facility USARs and substitute a description of the facility's participation in the ISP. For those facilities which are scheduled to remove a capsule as part of the ISP (*i.e.*, Dresden, LaSalle, and Peach Bottom), the proposed amendment would revise the material specimen withdrawal schedule in accordance with the ISP. Finally, for Oyster Creek, which is not scheduled to remove any further material specimens, the proposed amendment would revise the USAR to state that Oyster Creek will participate in the ISP.

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 adopts an integrated surveillance program (ISP) for reactor material specimen surveillances. The ISP ensures that the reactor pressure vessel (RPV) will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change. The initial conditions and methodologies used in accident analyses remain unchanged. The proposed change does not revise or alter the design assumptions for systems or components used to mitigate the consequences of accidents. Thus, accident analyses results are not affected by this proposed change.

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

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

The proposed change adopts an ISP for reactor material specimen surveillances. The ISP ensures that the RPV will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change.

The proposed change does not affect the design or operation of any system, structure, or component (SSC) in the plant. The safety functions of the related SSCs are not changed in any manner, nor is the reliability of any SSC reduced. The change does not affect the manner by which the facility is operated and does not change any facility, structure, system, or component.

No new or different type of equipment will be installed by this proposed change.

Therefore, the proposed amendment 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 has no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints. No physical or operational changes to the facility will result from the proposed changes. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

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

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) by: (1) Modifying the wording of the current Surveillance Requirements (SRs) 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2, Improved Standard Technical Specifications (ISTS) wording for SR 3.0.1 and SR 3.0.3; and (2) modifying the ISTS wording, adopted in item 1 above, 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 consistent with Technical Specification Task Force (TSTF)–358 for missed surveillances.

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). Entergy Operations Inc. reviewed the following proposed NSHC determination published in the **Federal Register** as part of the CLIIP for TSTF–358, and concluded in its application of August 19, 2002, that the proposed NSHC determination applied to Waterford Steam Electric Station, Unit 3.

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

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.

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.

3. 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.

Proposed Changes to SR 4.0.1 and 4.0.3

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 the revised SR 4.0.1 and 4.0.3 wording. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The proposed change involves rewording of the existing SRs 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. 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.

2. 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 SR 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.

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

The proposed change involves rewording of the existing SRs 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.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 16, 2002.

Description of amendment request: The proposed amendment will revise the current main steam isolation valve (MSIV) Technical Specification (TS) 3/4 7.1.5 to more closely reflect TS 3.7.2 contained in NUREG–1432, Revision 2. In addition, this change will remove the MSIVs from the scope of containment isolation valve (CIV) TS 3/4 6.3 such that only TS 3/4.7.1.5 will apply to the MSIVs. These changes will provide increased flexibility and clarity regarding the implementation of the TSs regarding MSIVs.

Basis for proposed no significant hazard 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 to the applicability for the main steam line isolation valves will not require operability when all MSIVs are closed in Modes 2, 3, and 4. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. In the closed position the MSIVs are already in their safety function position. In this position, there can be no increase in the probability or consequences of an accident.

The consequences of previously analyzed events are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. When the MSIVs are closed in Modes 2, 3, and 4 they are performing their design function for containment isolation and for main steam line isolation on the secondary side of the plant. The proposed change does not alter the initial conditions assumed in the safety analyses. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed change does not impose any new safety analyses limits. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed change increases the allowed outage time for an inoperable MSIV from 4 hours to 8 hours in Mode 1 and for Modes 2, 3, and 4; will allow both MSIVs to be inoperable, will allow separate action entry for the inoperable valves, and will allow 8 hours to close each inoperable valve. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Extending the time available to complete repairs of an inoperable component does not have a detrimental impact on the integrity of plant components nor does it increase the probability that these components will fail. The proposed changes are not related in any way to the probability of failure of a plant structure, system or component which would result in the occurrence of an analyzed event. Because the probability of failure of plant equipment is not affected, there is no impact on the probability of occurrence of a previously analyzed accident.

The consequences of previously analyzed events are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The steam line break analysis in FSAR [Final Safety Analysis Report] Section 15.1.3 assumes a failure of one MSIV to close. For the containment isolation function, in the event of an inoperable MSIV coincident with a LOCA [loss-of-coolant accident], the closed system (*i.e.*, the steam generator tubes and main steam line piping) remains intact. The closed system is subjected to a Type A containment leakage test, is missile protected, and [has] seismic category I piping, and typically has flow through it during normal operation such that any loss of integrity could be continually observed through leakage detection systems within containment and system walkdowns outside containment. Therefore, with an inoperable MSIV the safety analysis (both LOCA and steam line break) remains valid assuming no additional failures. The increase in core damage frequency and large early release fraction, resulting from the increased restoration time, is negligible. The proposed 8 hour Allowed Outage Time is sufficiently short to ensure that the MSIVs are operable when required to perform their design function. Even though both MSIVs will be allowed under separate condition entry, to be inoperable in Modes 2, 3, and 4 the inoperable valves are still required to be closed. The 8 hour Allowed Outage Time to close an inoperable valve is based on the small likelihood of an accident occurring that will need the MSIV isolation function during this time period and the fact that the valves are located on a closed system with respect to containment integrity. The proposed change does not alter the initial conditions assumed in the safety analyses. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed change does not impose any new safety analyses limits. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed change will add a Note to the MSIV surveillance to allow entry into Mode 3 for testing at hot conditions. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The addition of this allowance for testing is not related in any way to the probability of failure of a plant structure, system or component which would result in the occurrence of an analyzed event. Because the probability of failure of plant equipment is not affected, there is no impact on the probability of occurrence of a previously analyzed accident.

The consequences of previously analyzed events are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The proposed change will allow entry into Mode 3 in order to perform MSIV testing at hot conditions. However, prior to this testing, the MSIVs are not known to be inoperable from any other cause other than not having performed the Surveillance Requirement to demonstrate closure times at hot plant conditions, which they are expected to pass. The proposed change will allow entry into Mode 3 for the condition where both MSIVs may require closure time testing. This testing allowance is limited to Mode 3, and must be completed prior to entry into Modes 1 or 2. The proposed change does not alter the initial conditions assumed in the safety analyses. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed change does not impose any new safety analyses limits. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed change will require MSIVs, that are closed in accordance with the Mode 2, 3, and 4 Action, be verified closed once per seven days. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The addition of this requirement is not related in any way to the probability of failure of a plant structure, system or component which would result in the occurrence of an analyzed event. Because the probability of failure of plant equipment is not affected, there is no impact on the probability of occurrence of a previously analyzed accident.

The consequences of previously analyzed events are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event. The proposed change adds a Surveillance Requirement to Technical Specification 3/4.7.1.5 to verify proper MSIV isolation on an actuation signal. This is not a new Surveillance Requirement for the Technical Specifications. Technical Specification 3.3.2, Engineering Safety Features Actuation System Instrumentation, Surveillance Requirement 4.3.2.1 (Table 4.3–2 Item 4.d) requires a functional test of the actuation relay (K305) once per 18 months which verifies automatic closure of the MSIVs on a simulated main steam isolation signal. The proposed change does not alter the initial conditions assumed in the safety analyses. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed change does not impose any new safety analyses limits. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, none of the proposed change[s] described above involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant. 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, or to the setpoints at which protective or mitigative actions are initiated. 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.

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 margin of safety is established through equipment design, limitations on operating parameters, and the setpoints at which automatic actions are initiated. No equipment design features are impacted by this change, no operating parameters are revised, and no changes to the actuation setpoints are involved.

The design safety function of the MSIVs is to close upon receipt of a main steam isolation signal. With the MSIVs already closed in Modes 2, 3 or 4, the design function is satisfied.

The proposed change will increase the allowed outage time from 4 hours to 8 hours in Mode 1, for an inoperable MSIV. The proposed change will also relax current allowances for MSIVs in Modes 2, 3, and 4; however, the relaxations are in lower modes of operation where the potential for an accident that would require the MSIV isolation function is reduced. The proposed changes will still ensure that the inoperable MSIV(s) are restored or closed in a reasonable time of 8 hours. Once closed, the MSIVs meet their design safety function.

The proposed change will add a note indicating the Surveillance Requirements must be performed prior to entry into Modes 1 or 2. The MSIVs are expected to pass the Surveillance Requirement and are not known to be inoperable for any other reason than not having performed the valve closure test at hot conditions. The testing is limited to Mode 3, when the reactor is subcritical, thus verifying the MSIV closure times prior to power operation.

The proposed change will require MSIVs, which are closed in accordance with the Mode 2, 3, and 4 Action, be verified closed once per seven days. This requirement provides additional assurance that the MSIVs perform their design safety function to close.

The proposed change adds a Surveillance Requirement to Technical Specification 3/4.7.1.5 to verify proper MSIV isolation on an actuation signal. This, however, is not a new Surveillance Requirement for the Technical Specifications. Technical Specification 3.3.2, Engineering Safety Features Actuation System Instrumentation, Surveillance Requirement 4.3.2.1 (Table 4.3–2 Item 4.d) requires a functional test of the actuation relay (K305) once per 18 months which verifies automatic closure of the MSIVs on a simulated main steam isolation signal.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 16, 2002.

Description of amendment request: The proposed amendment will add the topical report entitled ''Fuel Rod Maximum Allowable Gas Pressure,'' CEN–372–P–A, to the list of analytical methods in Technical Specification (TS) 6.9.1.11.1 used to determine the Waterford Steam Electric Station, Unit 3 (Waterford 3) core operating limits.

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

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

Response: No.

The proposed change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident. The proposed change adds an NRC [Nuclear Regulatory Commission]-approved topical report to the list of analytical methods used to determine the core operating limits. The effect of the addition of this new reference is to revise the fuel design criterion for internal rod pressure to accept rod pressures that may exceed nominal Reactor Coolant System operating pressure. The use of this revised criterion continues to ensure that the consequences of an accident remain within acceptable limits. The change also proposes the administrative deletion of report date and revision levels in the list of references. These changes do not alter any of the assumptions or bounding conditions currently in the Final Safety Analysis Report.

Waterford 3 performed a large break loss-of-coolant accident (LOCA) analysis using bounding fuel performance data as described in CEN–372–P–A. This analysis concluded that the peak cladding temperature remained within 10 CFR 50.46 limits.

In addition to the LOCA analysis, an evaluation of the potential for departure from nucleate boiling (DNB) propagation was performed as described in CEN–372–P–A. The results confirmed that Waterford 3 is bounded by the results evaluated in the topical report and that DNB propagation will not occur.

Based on these analyses, there is no 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 change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed change. The intent of the proposed change is to reference an NRC-approved topical report in the Technical Specifications. The topical report justifies an acceptance criterion that allows fuel rod internal pressure to exceed RCS [reactor coolant system] pressure. There are no new accidents created by this change. An administrative aspect of this change, the deletion of date and revision levels, was also considered and does not create a new or different accident.

The impact of fuel rod internal pressure exceeding reactor coolant system (RCS) pressure was considered in both an emergency core cooling system (ECCS) performance analysis and in a DNB propagation evaluation performed for Waterford 3. These two aspects were required considerations based on the NRC Safety Evaluation review of the topical report. The results demonstrated that Waterford 3 continues to meet 10 CFR 50.46 and that there is no potential for DNB propagation.

Based on these analyses, there is no possibility of the creation of a new or different kind of accident from those previously evaluated.

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

Response: No.

The proposed change adds an NRC-approved topical report to the list of analytical methods used to determine core operating limits. It also deletes the revision number and dates associated with each of the topical reports listed. The effect of the addition of the new reference is to revise the fuel design criterion for fuel rod internal pressure to accept rod pressures that may exceed nominal RCS operating pressure. The use of this revised criterion continues to ensure that the consequences of an accident remain within acceptable limits. Since the core operating limits will continue to be established by an NRC-approved methodology and the results will be verified to meet the established acceptance criteria of 10 CFR 50.46, the change will provide adequate core protection. Thus, the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 20, 2002.

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

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

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

Response: No.

The proposed changes are primarily to correct titles, page numbering errors, and otherwise make the TS index pages consistent with other NRC [U. S. Nuclear Regulatory Commission] approved pages. These changes are all of an administrative nature and have no effect on any plant equipment or structures. Therefore, these changes do not increase the probability or consequences of an accident previously evaluated.

The proposed amendment also deletes TS 5.4.1 and 5.4.2. Values for RCS [Reactor Coolant System] design pressure, temperature, and volume are contained in the Final Safety Analysis Report. Any changes to these are controlled by 10 CFR 50.59. Therefore, removing the section from the TS will not increase the probability or consequences of previously evaluated accidents.

The proposed amendment also deletes TS 6.6 and 6.7, and revises TS 6.9.1 and TS 6.9.2 to administratively conform reporting requirements to those in 10 CFR [part] 50. Therefore, removing these sections from the TS will not increase the probability or consequences of previously evaluated accidents.

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

Response: No.

The proposed changes are administrative in nature and do not involve a physical alteration of the plant. No new or different equipment or modes of operation are being introduced by this proposed change. Thus, the 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.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes are primarily administrative in nature and can not affect any safety barriers. The proposed change to TS 5.4 only deletes unnecessary information. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 6, 2002.

Description of amendment request: The proposed amendment would increase the surveillance interval of the Local Power Range Monitor (LPRM) calibrations from 1000 megawatt-days/ton to 2000 megawatt-days/ton.

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:

Operation of the JAF [James A. FitzPatrick] plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

1. Involve an increase in the probability or consequences of an accident previously evaluated. The revised surveillance interval continues to ensure that the LPRM signal is adequately calibrated. The proposed change results in no change in radiological consequences of the design basis LOCA [loss-of-coolant accident] as currently analyzed for JAF. This change will not alter the basic operation of process variables, structures, systems, or components as described in the JAF UFSAR [Updated Final Safety Analysis Report], and no new equipment is introduced by the change in LPRM surveillance interval. The performance of the APRM [Average Power Range Monitor] and RBM [Rod Block Monitor] systems are not significantly affected by the proposed LPRM surveillance interval increase. Therefore, the probability of accidents previously evaluated is unchanged.

The consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident, but LPRM chamber exposure has no significant effect on the calculated thermal limits because LPRM accuracy does not significantly deviate with exposure. For the extended calibration interval, the total nodal power uncertainty remains less than the uncertainty assumed in the thermal analysis basis safety limit, maintaining the accuracy of the thermal limit calculation. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency, and the consequences of an accident previously evaluated are unchanged.

The change does not affect the initiation of any event, nor does it negatively impact the mitigation of any event. Therefore, the proposed change does 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 change will not physically alter the plant or its mode of operation. The performance of the APRM and RBM systems are not significantly affected by the proposed LPRM surveillance interval increase. 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 are consistent with current safety analysis assumptions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety. The proposed change has no impact on equipment design or fundamental operation, and there are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed change. The performance of the APRM and RBM systems are not significantly affected by the proposed LPRM surveillance interval increase. The margin of safety can be affected by the thermal limits existing prior to an accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. The thermal limit calculation is not significantly affected because LPRM sensitivity with exposure is well defined. LPRM accuracy remains within the total nodal power uncertainty assumed in the thermal analysis basis, thus maintaining thermal limits and the safety margin.

Since the proposed change does not affect safety analysis assumptions or initial conditions, the margin of safety in the safety analyses are maintained. 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: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: January 9, 2003.

Description of amendment request: The proposed Technical Specification (TS) amendment request changes the definition of a Logic System Functional Test, deletes the definition of a Simulated Automatic Actuation, clarifies Surveillance Requirement 4.5.G.1.a regarding simulated automatic actuation testing, and revises associated TS Bases.

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

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

The proposed change involves surveillance requirements and definitions of surveillance tests. As such, the proposed change does not involve any plant physical changes, change any Technical Specification instrumentation setpoints, or introduce any new mode of plant operation. The proposed change to surveillance requirements and definitions does not result in any significant change in the availability of logic systems or safety-related systems themselves. Protective functions will be maintained. The proposed change does not degrade plant design, operation, or the performance of any safety system assumed to function in the accident analysis.

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

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

The proposed change does not: introduce any new accident initiators or failure mechanisms because the changes do not introduce any new modes of plant operation, make any physical changes (no new or different type of equipment will be installed); or change any Technical Specification instrumentation setpoints or methods of plant operation. The proposed changes will not substantially impose new requirements or eliminate any existing requirements.

Therefore, the changes to the surveillance requirements and testing definitions that encompass this proposed change do not create the possibility of a new or different kind of accident than those previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There is no change or impact on any safety analysis assumptions. The proposed change does not involve any increase in calculated off-site dose consequences. Operability of protective instrumentation and the associated systems is unaffected, and performance of equipment will not be significantly affected. Since the proposed change is consistent with the BWR/4 Standard Technical Specifications, NUREG–1433, Revision 2, approved by the NRC [Nuclear Regulatory Commission] staff, revising the Technical Specifications in a manner which clarifies and reflects the approved level of detail ensures that safety margins are acceptable. Therefore, there is no significant reduction in the margin of safety as a result of this Technical Specification change.

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. Clifford.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: March 14, 2002.

Description of amendment request: The proposed license amendment request (LAR) will allow exercising and testing the Inclined Fuel Transfer System (IFTS) prior to the beginning of the refueling outage, thus increasing system reliability and refuel outage efficiency. The proposed LAR does not provide for the movement of fuel. The proposed LAR supplements Amendment No. 100 by including a time limit on the removal of the IFTS blind flange, providing a requirement to install the upper pool IFTS gate prior to IFTS blind flange removal, and limiting the unbolted configuration on the IFTS blind flange when it is rotated.

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 permits removal of the Inclined Fuel Transfer System (IFTS) blind flange for a maximum duration of 60 days per cycle when primary Containment operability is required in MODES 1 (Power Operation), 2 (Startup), or 3 (Hot Shutdown). The proposed change also limits the duration the IFTS blind flange may be unbolted when in MODES 1, 2, or 3. The proposed change does not involve modifications to plant systems or design parameters that could contribute to the initiation of any accidents previously evaluated.

Regarding the probability and consequences of design basis and beyond design basis accidents, a comprehensive technical evaluation was completed in accordance with Regulatory Guide (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis'' and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications.'' This evaluation determined that the proposed change is technically justified and the associated risk is insignificant.

The proposed change permits alteration of the containment boundary for the IFTS penetration. Regarding the consequences of accidents, the proposed change has been determined via a probabilistic risk assessment to be acceptable regarding its overall impact to the plant's risk, consistent with the Nuclear Regulatory Commission's Safety Goal Policy Statement. The resulting pressures and temperatures from a design basis Loss Of Coolant Accident (LOCA) are considered the primary challenge to the integrity of the containment. Pursuant to Amendment 100, the existing Technical Specifications require maintaining an adequate water seal to prevent leakage from the bottom of the IFTS transfer tube and isolating the drain piping. This water seal is adequate to mitigate the effects of the design basis peak post-accident pressures and temperatures. The proposed change requires the installation of the upper IFTS pool gate to provide protection of the Suppression Pool Make Up system water inventory. A time limit for IFTS blind flange removal of 60 days per cycle and a 20 hour limit for the unbolted configuration of the IFTS flange have been established as conservative measures to limit the associated risk to the containment boundary for all accident conditions. The proposed change has been found to be acceptable regarding flooding and seismic design issues.

Therefore, the function of the containment to provide an adequate boundary in the event of a design basis LOCA is not compromised with the proposed change and the proposed change does not result in a significant increase in the probability of the consequences of previously evaluated accidents.

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

The proposed change consists of the removal of the IFTS blind flange when in MODES 1, 2, or 3. The IFTS blind flange is a passive component that is not part of the primary reactor coolant pressure boundary and is not involved in the operation or shutdown of the reactor. Being passive, its presence or absence does not affect any of the parameters or conditions that could contribute to the initiation of any incidents or accidents that are created from a loss of coolant or positive reactivity incident. Re-aligning the boundary of the primary containment to include portions of the IFTS is passive in nature and therefore has no influence on the possibility of creating a new or different kind of accident. Furthermore, operation of the IFTS is unrelated to the operation of the reactor and there is no mishap in the process that can lead or contribute to the possibility of losing any coolant in the reactor or introducing the chance for positive or negative reactivity or other accidents different from and not bounded by those previously evaluated.

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

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

The proposed change involves the re-alignment of the primary containment boundary by removing the IFTS blind flange, which is a passive component. The margin of safety that has the potential of being impacted by the proposed change involves the dose consequences of postulated accidents, which are directly related to potential leakage through the primary containment boundary. The potential leakage pathways due to the proposed change have been reviewed, and leakage can only occur from the administratively controlled IFTS transfer tube drain piping. Pursuant to Amendment 100, an individual is currently designated to provide timely isolation of this drain piping when this proposed change is in effect. The conservatively calculated dose, which might be received by the designated individual while isolating the drain piping, is well within the guidelines of General Design Criterion 19. Furthermore, the drain piping isolation valve is included in the Primary Containment Leakage Rate Testing Program to ensure that leakage from the piping and components located outboard of the blind flange will be maintained consistent with the leakage rate assumptions of the accident analysis. It has been determined that the proposed change would not have a substantial impact on the ultimate pressure capacity of the containment as it relates to the Large Early Release Frequency (LERF) nor would it have a substantial impact on LERF from seismic events. Therefore, the dose consequences of an event would be unchanged, and the associated margin of safety would also be unchanged.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: March 14, 2002.

Description of amendment request: The amendment request proposes a one-time exception to the requirement in Nuclear Energy Institute (NEI) 94–01 to perform an integrated leak rate test (ILRT) at a frequency of 10 years. The exception is to allow ILRT testing within 15 years from the last ILRT, completed July 1, 1994. The proposed amendment is considered risk-informed, therefore Regulatory Guide 1.174, ''An approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' has been followed, while using the methodology of Electric Power Research Institute (EPRI) report, ''Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,'' (EPRI TR–104285).

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

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurence, nor is it an activity or modification that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that reducing Type A (ILRT) testing frequency to once per twenty years leads to an imperceptible increase in risk.

Other testing and inspections provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The last three Type A tests performed at PPNP identified containment leakage within the acceptable criteria, indicating a very leak-tight containment. Inspections required by the ASME Code are performed in order to identify indications of containment degradation that could affect leak-tightness. Containment pressure is monitored each shift during plant operation and would identify containment vessel shell leakage into the annulus by a decrease in containment pressure. Type B and C testing, required by Technical Specifications, identifies any containment leakage from designed penetrations, such as from valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the PPNP Type A test interval will not represent a significant increase in the consequences of an accident.

Thus, the proposed amendment does not involve a significant increase in the probability or consequences of a previously evaluated accident.

2. This proposed amendment 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 adds a one-time extension to the current interval for Type A testing for PPNP. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes to the plant or changes to the operation of the plant that could introduce a new failure.

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

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

The proposed revision to the PPNP Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes only about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal effect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. Furthermore, for PPNP, monitoring containment vessel pressure each shift during operation further reduces the risk of any containment leakage path going undetected. The PPNP test and inspection performance has satisfactorily demonstrated that the containment remains very leak tight. The proposed change has no effect on Core Damage Frequency (CDF). The change in Large Early Release Frequency (LERF) was computed and found to be a ''very small'' change in accordance with the guidelines of Regulatory Guide 1.174. The computed change in Conditional Containment Failure Probability (CCFP) and offsite dose have also been evaluated and are considered to be insignificant.

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

Based on the above considerations, it is concluded that a significant hazard would not be introduced as a result of this proposed change.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would revise Crystal River Unit 3 Improved Technical Specifications (ITS) 3.3.15 ''Reactor Building Purge Isolation-High Radiation;'' ITS Bases 3.7.15 ''Spent Fuel Assembly Storage;'' ITS 3.9.3 ''Containment Penetrations;'' and ITS 3.9.6 ''Refueling Canal Water Level'' to account for handling irradiated fuel within containment that has not 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:

Crystal River Unit 3 (CR–3) proposes to revise Improved Technical Specifications (ITS) 3.3.15, 3.9.3, 3.9.6, and Bases 3.7.15.

Florida Power Corporation (FPC) has determined that this license amendment request does not involve a significant hazards consideration as defined in 10 CFR 50.92 based on the following:

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

The proposed change does not increase the probability of a fuel handling accident in that the proposed change deals with the results of such an accident, not the cause of such an accident. The proposed change does not increase the consequences of an accident previously evaluated in that the CR–3 Alternate Source Term (AST) has been approved by the NRC, and this proposed change implements that NRC approval. The AST for the Fuel Handling Accident (FHA) takes no credit for containment isolation nor for a filtered release.

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

The proposed changes to the ITS do not affect nor create a different type of fuel handling accident. The fuel handling accident analyses assume that all of the iodine and noble gases that become airborne, escape, and reach the exclusion area boundary and low population zone with no credit taken for filtration, containment of the source term, or for decay or deposition in the containment. The proposed changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. The revised operations are consistent with the fuel handling accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The calculated doses to both the public and control room operators are well within the limits given in 10 CFR 50.67. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any ITS that is related to the establishment of or maintenance of a safety margin.

The systems that have been included in the proposed change will have administrative controls in place to assure that the systems are available and can be promptly returned to operation to further reduce dose consequences. These administrative controls will include a single normal or contingency method to promptly close the equipment hatch opening. This prompt method need not completely block the hatch opening nor be capable of resisting pressure, but is to enable the ventilation systems to draw the release from the postulated FHA in the proper direction such that it can be monitored. Therefore, operations of the facility in accordance with the proposed amendment would not involve a significant reduction in margin of safety.

Based on the above, FPC concludes that the proposed license 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: R. Alexander Glenn, Associate General Counsel (MAC-BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 19, 2002.

Description of amendment request: The proposed amendment would revise Crystal River Unit 3 Improved Technical Specification 2.1.1, ''Reactor Core Safety Limits.'' The proposed change will permit the use of the BHTP correlation, which is needed to utilize the Framatome ANP high thermal performance (HTP) spacer grid design.

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

FPC [Florida Power Corporation] has evaluated the proposed License Amendment Request (LAR), which consists of the identified Improved Technical Specification (ITS) change, against the criteria of 10 CFR 50.92(c). The ITS change allows the use of the BHTP Correlation for departure from nucleate boiling (DNB) calculations of reload cores containing the Mark-B/HTP fuel design.

FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the criteria is satisfied.

(1) [Does not] [i]nvolve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed safety limit value ensures that fuel integrity will be maintained during normal operations and anticipated operational occurrences (AOOs), and that the design requirements will continue to be met. The proposed methodology for the BHTP departure from nucleate boiling (DNB) correlation will be generically reviewed and approved by the NRC prior to its use by Crystal River Unit 3 (CR–3) in mixed core reload analyses. The core operating limits will be developed in accordance with the new methodology and any limitations established by the NRC in its safety evaluation of the new methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. Therefore, the safety limit value for the BHTP correlation will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) [Does not] [c]reate the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident/event initiator. No new initiating events or transients result from the use of the BHTP correlation and the related safety limit changes. Therefore, the safety limit value for the BHTP correlation will not involve the possibility of a new or different kind of accident from any previously evaluated.

(3) [Does not] [i]nvolve a significant reduction in a margin of safety.

The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation, to ensure that the applicable margin of safety is maintained (*i.e.*, there is at least 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience departure from nucleate boiling (DNB)). The proposed methodology for the BHTP DNB correlation will be generically reviewed and approved by the NRC prior to its use by CR–3. The other reactor core safety limits will continue to be met by analyzing the reload for the mixed core using NRC approved methods, and incorporation of resultant operating limits into the Core Operating Limits Report (COLR). Therefore, the safety limit value for the BHTP correlation 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: R. Alexander Glenn, Associate General Counsel (MAC-BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

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: December 20, 2002.

Description of amendment request: This proposed amendment provides editorial and administrative changes to the Technical Specifications. The changes correct typographical, spelling, numbering syntax, page break, and font consistency errors as well as removing blank pages and associated references. There are no substantive changes made in the proposed amendment.

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 proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments are purely administrative or editorial in nature. These amendments make no substantive Technical Specification changes and do not affect any assumptions contained in plant safety analyses, the physical design and/or operation of the plant; and they do not affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

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 use of the administratively changed Technical Specifications does not create the possibility of a new or different kind of accident from any previously evaluated, since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes and clarifications, since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

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

No. The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The changed Technical Specifications, which correct administrative and editorial errors, and clarify existing Technical Specification requirements, do not reduce any of 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 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.

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

Date of amendment request: December 31, 2002.

Description of amendment request: The proposed amendment would change the reactor vessel material surveillance program to incorporate the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) into the licensing basis.

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.

Pressure-temperature (P/T) limits (CNS [Cooper Nuclear Station] Technical Specifications Figures 3.4.9–1, 2, and 3) are imposed on the reactor coolant system to ensure that adequate safety margins against non-ductile or brittle fracture exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P/T limits are based on the nil-ductility reference temperature, RT_{NDT}, as described in ASME Section XI, Appendix G. Changes in the fracture toughness properties of RPV [reactor pressure vessel] beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR 50 [title 10 of the Code of Federal Regulations part 50] Appendix H. The effect of neutron fluence on the shift in the RT_{NDT} of RPV materials is predicted by methods given in RG [Regulatory Guide] 1.99, Revision 2.

This change is not related to any accidents previously evaluated. Rather, the reactor vessel surveillance program, corresponding material evaluations, and adjustment of a plant's P/T limits, as necessary, protect against the possibility of reactor vessel brittle fracture. Monitoring, evaluation, and adjustment of CNS P/T limits to ensure adequate margin exists to brittle fracture will continue. This change only replaces a plant-specific monitoring and evaluation program with an integrated industry program, the BWRVIP ISP. The NRC has reviewed this program and approved it for implementation in a Safety Evaluation, dated February 1, 2002.

CNS's current P/T limits were established based on adjusted reference temperatures developed in accordance with the procedures described in RG 1.99, Revision 2. Calculation of adjusted reference temperature by these procedures includes a margin term to ensure conservative, upper-bound values are used for the calculation of the P/T limits. This change does not affect the existing P/T limits in the CNS Technical Specifications Figures 3.4.9–1, 2, and 3. This change will not affect any plant safety limits or limiting conditions of operation. The proposed change will not affect reactor pressure vessel performance as no physical changes are involved aside from changes related to surveillance capsule withdrawal, and CNS vessel P/T limits will remain conservative in accordance with RG 1.99, Revision 2 criteria. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological 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 revises the CNS license basis to reflect participation in the BWRVIP ISP. Participation in the BWRVIP ISP will continue to ensure that the CNS reactor vessel materials are monitored and evaluated as necessary to protect against brittle fracture. This proposed change does not involve a modification of the design of plant structures, systems, or components. The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of structures, systems, or components important to safety as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of equipment will not be increased, and increased or more severe testing of equipment will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from that previously evaluated.

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

Response: No.

Conformance with 10 CFR [part] 50 Appendix G defines the accepted safety margin for Reactor Coolant Pressure Boundary fracture toughness. The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor pressure vessel, a condition that is unanalyzed. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

This proposed change will not alter the required margins as defined in 10 CFR [part] 50, Appendix G. This proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed change does not involve revision of the P/T limits. Rather, this change involves a revision to the surveillance capsule withdrawal schedule, a revision to the reactor vessel fluence calculational methodology to achieve consistency within the BWRVIP ISP, and participation in future BWRVIP ISP developments. The current P/T limits were established based on adjusted reference temperatures for vessel beltline materials calculated in accordance with RG 1.99, Revision 2 which will continue to conform to 10 CFR [part] 50 Appendix G. Therefore, the proposed change does not involve a significant reduction in any safety margins.

In summary, it is concluded that this License Amendment Request does not involve significant hazards consideration results. NPPD has researched the existing regulatory precedent and has identified five BWR licensees with similar License Amendment Requests currently under NRC staff review:

- Browns Ferry Units 2 and 3—Submittal date November 6, 2002.
- Monticello Generating Station—Submittal date September 19, 2002.
- River Bend—Submittal date August 15, 2002.
- Fermi Unit 2—Submittal date August 8, 2002.
- Susquehanna Units 1 and 2—Submittal date July 25, 2002.

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. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 13, 2003.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) operating license and Technical Specifications to increase the licensed rated power by 1.4 percent to 1673 megawatts thermal (MWt) using measurement uncertainty recapture.

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 Kewaunee Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

There are no changes as a result of the measurement uncertainty recapture (MUR) power uprate to the design or operation of the plant that could affect system, component, or accident mitigative functions. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

The reduction in power measurement uncertainty allows for some of the safety analyses to continue to be used without modification. This is because the safety analyses were performed or evaluated at either 102 percent of 1650 MWt or higher. Analyses at these power levels support a core power level of 1673 MWt with a measurement uncertainty of 0.6 percent. Radiological consequences of USAR [updated safety analysis report] chapter 14 accidents were assessed previously using the alternate source term (AST) methodology (reference 7.1, TAC [technical assignment control] No. MB4596). These analyses were performed at 102 percent of 1650 MWt and continue to be bounding. The USAR chapter 14 analyses and accident analyses submitted to the NRC [Nuclear Regulatory Commission] with the fuel transition (reference 7.3, TAC No. MB5718) continue to demonstrate compliance with the relevant accident analyses acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) were evaluated at an uprated core power level of 1772 MWt and continue to comply with their applicable structural limits. These analyses also demonstrate the components will continue to perform their intended design functions. Changing the applicability of the heatup and cooldown curves is based on uprated fluence values. This does not have a significant effect on the reactor vessel integrity. Thus, there is no significant increase in the probability of a structural failure of the primary loop components.

All of the NSSS [Nuclear Steam Supply System] systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/BOP [balance of plant] interface systems were evaluated at 1772 MWt and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety-related systems or component and does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

Operation at the 1673 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a two percent power measurement uncertainty or at uprated core powers that exceed the MUR uprated core power. System and component analyses have been completed at a core power in excess of the MUR uprated core power. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been either reviewed and approved by the NRC, are in the process of being approved by the NRC, or are in compliance with applicable regulatory review guidance and standards. 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: John H. O'Neill, Jr., Esq., Shaw Pittman, Potts & Trowbridge, 2300 N. Street, NW., Washington, DC 20037–1128.

NRC Section Chief: L. Raghavan.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: September 12, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' and TS 3.5.3, ''ECCS-Shutdown,'' to add a surveillance requirement to verify every 31 days that the ECCS piping is full of water; consistent with NUREG–1431, Standard Technical Specifications, Westinghouse Plants, Revision 2.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

This license amendment request proposes to add a surveillance requirement to verify the ECCS is full of water every 31 days while operating in Modes 1, 2, 3 and 4.

This proposed change does not cause an increase in the probabilities of any accidents previously evaluated, because the change will not cause an increase in the probability of any initiating events for accidents previously evaluated. In particular, the change affects the ECCS, which serves to mitigate rather than initiate accidents.

The consequences of the accidents previously evaluated in the PBNP [Point Beach Nuclear Plant] Final Safety Analysis Report (FSAR) are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems. The change proposed in this license amendment request provides an appropriate surveillance requirement for the ECCS, and thus does not increase the probability of failure of this equipment or its ability to operate as required for the accidents previously evaluated in the PBNP FSAR.

Therefore, the consequences of an accident previously evaluated in the PBNP FSAR will not be significantly increased as a result of the proposed change, because the factors that are used to determine the consequences of accidents are not being changed.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

Equipment important to safety will continue to operate as designed. The proposed change does not result in any event previously deemed incredible being made credible. The change does not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendment will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries and will not cause a release of fission products to the public. Venting the piping associated with a train of ECCS will render that ECCS train inoperable while it is being vented. Performance of this surveillance will therefore affect the availability of the associated ECCS train, but performance of the surveillance requirement at the specified frequency is consistent with the requirements of NUREG–1431, Standard Technical Specifications for Westinghouse Plants, Revision 2. Additionally, verifying the ECCS piping is full of water ensures that the system will perform properly, injecting its full capacity into the RCS [reactor coolant system], upon demand. Therefore, adopting a surveillance requirement to verify the ECCS piping is full of water, will not result in more than a minimal 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: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: September 26, 2002.

Description of amendment request: The proposed change would revise the Steam Generator low-low level trip setpoint and allowable values provided in the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications Table 2.2–1, ''Reactor Trip System Instrumentation Trip Setpoints,'' and Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation Trip Setpoints.'' The changes are necessary based on PSEG Nuclear's evaluation of a loss of feedwater transient at Diablo Canyon. During the event, Diablo Canyon personnel observed a flow induced pressure drop in the steam generator mid-deck area. The proposed change accounts for a level measurement bias resulting from the pressure drop that was not considered in the previous Westinghouse analysis. This bias has the effect of providing nonconservative level readings and setpoints.

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

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

The proposed change to Tables 2.2–1 and 3.3–4 changes both the allowable trip setpoint and allowable value for the Steam Generator Water Level-Low-Low from ≥9.0% to ≥14.0% and from ≥8.0% to ≥13% respectively. The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The signal is used as a primary protection signal for the design basis loss of normal feedwater, loss of offsite power and feedwater line break safety analysis. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The change in the setpoint and allowable value allows the trip to function as originally designed accounting for the differential pressure created by steam flow past the mid-deck plate in the moisture separator section of the steam generator.

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

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

The proposed changes to the Steam Generator Water Level-Low-Low trip setpoint and allowable values allow the trip to function as originally designed. They do not alter the plant configuration in any way, and do not replace or modify existing plant equipment, or affect any plant operations. No additional failure mechanisms are introduced as a result of the changes to the setpoints and allowable values.

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

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

The proposed changes to the allowable trip setpoint and allowable value for the Steam Generator Water Level-Low-Low trip maintains core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity.

Therefore, it is concluded that the proposed changes to the steam generator low low level trip setpoint and allowable value[s] do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 23, 2002.

Description of amendment request: The proposed change would revise the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specification (TS) 6.12, ''High Radiation Area'' to be consistent with the Standard TSs for Westinghouse Plants (NUREG–1431, Revision 2) by updating the current reference to title 10 of the Code of Federal Regulations (10 CFR), section 20.203 with the corresponding reference to 10 CFR 20.1601.

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

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

The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility, or manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended safety function to mitigate the consequences of an initiating event within the acceptance limits assumed in the UFSAR. The proposed changes are administrative in nature. Technical Specification (TS) 6.12 will be updated to include the new 10 CFR 20 (effective 06/20/91) requirements. The proposed changes do not alter the conditions or assumptions in any of the previous accident analyses, and as a result, the radiological consequences associated with these analyses remain unchanged.

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

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated.

The proposed changes are administrative in nature and the relocated procedural details do not change the level of programmatic controls and procedural details. Accordingly, the proposed changes do not create any new failure modes or limiting single failures associated with a plant structure, system, or component important to safety. Also, there will be no change in the types or increase in the amounts of any effluents released offsite.

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

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

The proposed changes do not impact equipment design or operation, nor do the changes affect any TS safety limits or safety system settings that could adversely affect plant safety. The proposed changes are administrative in nature. Technical Specification (TS) 6.12 will be updated to include the new 10CFR20 requirements (effective 06/20/91) and are in conformance with NUREG–1431. Furthermore, the proposed changes do not result in a change in the types or an increase in the amounts of any effluents released offsite.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 4, 2002 (TS 02–07).

Description of amendment request: The proposed amendment would revise Technical Specification 6.8.4.h, ''Containment Leakage Rate Testing Program,'' to allow a one-time, 5-year extension to the current 10-year test interval for the performance-based leakage rate test program for 10 CFR 50, Appendix J, Type A 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change for extending Type A test frequency does not significantly increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident. TVA performed an evaluation of the risk significance for the proposed increase to the SQN Units 1 and 2 Type A test frequency. The results of the TVA risk evaluation indicates that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in NRC Regulatory Guide (RG) 1.174, ''An Approach for Using Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.'' TVA's evaluation indicates that the increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population is also non-risk significant. The proposed test interval extension does not involve a significant increase in the consequences of an accident. Research documented in NUREG–1493 determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C testing provides assurance that containment leakage from penetration leak paths (*i.e.*, valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests. In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. In addition to the IWE examinations, TVA will perform additional nondestructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas) at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5-year extended interval. Accordingly, TVA's proposed extension of the Type A test interval does not increase the probability or consequences of an accident previously evaluated.

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

The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident because there are no physical changes made to the plant or plant equipment governing normal plant operation. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident. TVA will perform additional non-destructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas) at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5 year extended interval.

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

The proposed change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG–1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal affect on risk because most potential leakage paths are detected by Type C testing. Previous Type A leakage tests conducted on SQN Units 1 and 2 indicate that leakage from containment have been less than the 10 CFR 50, Appendix J leakage limit of 1.0 L_a. A review of the previous Type A test results indicate a stable trend with a 10 percent margin below the 1.0 L_a leakage limit. Accordingly, these test results, in conjunction with the research findings from NUREG–1493, provide assurance that the proposed extension to the Type A test interval would not significantly reduce the margin of safety. Based on the above, TVA 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of application for amendments: November 15, 2002 (TS 02–06).

Brief description of amendments: The proposed amendments would revise the Technical Specification (TS) 3.7.1.3, ''Condensate Storage Water,'' Limiting Condition for Operation by increasing the required minimum amount of stored water from 190,000 gallons to 240,000 gallons. This change is being made to support the replacement steam generator requirements.

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

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

The proposed change does not change the physical design and construction of the condensate storage tank (CST). The purpose of the increased water volume is to ensure that the required volume of water, preserved by the technical specification (TS), is sufficient to meet Sequoyah Nuclear Plant (SQN) Licensing and Design Basis after installation of the replacement steam generators. The change in the administratively controlled inventory of the CST will not increase the probability of an accident. Therefore, the proposed change does not involve a significant increase in the probability of 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?

This change increases the minimum required volume of water in the CST, thus ensuring that the auxiliary feedwater (AFW) system can perform its required safety function, using a preferred water source for plant transient mitigation. The maximum and normal water levels in the CST are not being changed. Additionally, increasing the minimum water volume requirement will not initiate any accident. 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?

This change does not reduce any margin associated with the CST inventory available to AFW. The requirement for sufficient CST volume to maintain hot standby and subsequent cooldown to hot shutdown continues to be met by the minimum volume increase. Additionally, the essential raw cooling water (ERCW) system still provides the long-term supply of safety grade cooling water to the AFW in the event that all inventory of the CST is lost. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: November 15, 2002 (TS 02–01).

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to revise the trip setpoint column of the Reactor Protection System and Engineered Safety Features (ESF) instrumentation tables to utilize a nominal setpoint value and revise the associated Bases discussions.

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

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

The proposed revisions for the nominal trip setpoint representation are administrative changes that will not impact the application of the reactor trip or ESF actuation system instrumentation requirements. This is based on the setpoint requirements being applied without change, as well as the Avs [allowable values], in accordance with the setpoint methodology. The removal of the inequalities associated with the trip setpoint values will be more appropriate for the use of nominal setpoint values but will not differ in application from the setpoint methodology utilized by TVA. The revision of the radiation monitoring instrumentation table to use an Av will continue to provide appropriate operability limits. Deletion of the nominal terminology associated with overtemperature delta temperature average temperature at rated thermal power (T') and reactor coolant system power operated relief valve (PORV) lift settings provides a better representation of the limits associated with these values. In addition, this change will not alter plant equipment or operating practices. Therefore, the implementation of these changes will not increase the probability or consequences of an accident.

The revision of the reactor coolant pump (RCP) underfrequency trip setpoint and the Avs for the RCP underfrequeny and undervoltage and the containment purge radiation high has been evaluated and the results are documented in approved calculations. These calculations verify that the revised values are acceptable in accordance with appropriate calculation methodologies and that they will continue to support the accident analysis. This is based on margin being available in the accuracy determinations that could be used without impacting the intended functions of this instrumentation and maintains the established safety limits. These revisions will not require changes to the instrumentation settings currently being used or the methods for maintaining them. The offsite dose potential will not be impacted because this instrumentation will continue to adequately provide the designed safety functions to limit the release of radioactivity. Therefore, the proposed revision of these values will not significantly increase the probability or consequences of an accident.

The relocation and enhancement of current radiation monitoring and loss of voltage functions to new LCOs [limiting condition for operations] does not alter the intended functions of these systems or physically alter these systems. While some requirements have change[d] from current limitations, these changes have provided more appropriate criteria to ensure that the accident mitigation functions are maintained properly and are available. Changes to Avs have been evaluated in accordance with TVA setpoint methodology and have been verified to acceptably protect the associated safety limits. Format changes provide a clearer representation of the requirements and provide more consistency with the standard TSs in NUREG–1431. These changes continue to support or improve the required safety functions and therefore, will not increase the possibility or consequence of an accident.

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

The revision of the nominal trip setpoint representation and elimination of the nominal nomenclature, as well as the revised setpoint value and Avs, and the relocated LCOs will not alter the plant configuration or functions. The revised setpoint and the proposed operability limits will continue to provide acceptable initiation of safety functions for the mitigation of postulated accidents as required by the design basis. The primary function of the reactor protection system, the ESF actuation system, and the new actuation function LCOs is to initiate accident mitigation functions. These functions are not considered to be initiators of postulated accidents. The PORVs provide accident mitigation functions and could be the source of a loss of coolant accident. However, a clarification of how to apply the actuation setpoints without a change to the setpoints will not impact accident generation. The proposed changes do not create the possibility of a new or different kind of accident because the design functions are not altered and the proposed values meet the accident analysis requirements for accident mitigation.

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

The setpoint and Av revisions proposed in this request were evaluated and found to be acceptable based on operating margin available in the accuracy determinations. The reassignment of this excess margin to the setpoint and Av will not impact the safety limits required for the associated functions. The nominal trip setpoint representation change and the elimination of inappropriate nominal indications does not alter the TS functions or their application and will not require changes to design settings. The relocated requirements to new LCOs provide appropriate limits and enhancements to the actuation functions. Plant systems will continue to be actuated for those plant conditions that require the initiation of accident mitigation functions. The margin of safety is not significantly reduced because the proposed changes to the Av and setpoint representations will not change design functions and the initiation of accident mitigation functions for appropriate plant conditions will not be adversely impacted.

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

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

NRC Section Chief: Allen G. Howe.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: November 5, 2002.

Description of amendment request: The proposed changes would delete the monthly analog rod position test for the control rod bottom bistables currently required by Technical Specification (TS) Table 4.1–1, Item 9.

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

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

The proposed change deletes the monthly analog rod position test that verifies the operation of the rod bottom bistables. However, the TSs still require bistable action to be functionally verified to ensure operability on an 18-month frequency as part of the overall analog rod position indication system calibration. Furthermore, the TS-required monthly rod bottom bistable action test was being performed to address instrument drift in the rod bottom setpoint, which will essentially be eliminated by the design of new digital-based IRPI [Individual Rod Position Indication] electronics being installed. Consequently, elimination of the monthly rod bottom bistable action test will not result in the failure of any plant structures, systems, or components and does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated is not significantly increased.

Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. Deletion of the monthly rod bottom bistable action test does not affect the ability of the control rods to perform their function. Surveillance tests to verify the operability of the IRPI System are still being performed. Furthermore, the Rod Position Demand Counter System provides redundant control rod position indication. As a result, the consequences of any accident previously evaluated are not significantly affected by the proposed change.

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

The proposed change deletes the monthly surveillance of rod bottom bistable action in the Individual Rod Position Indication system. This change does not alter the methods governing normal plant operation. The IRPI provides indication of rod position, is one of two independent systems that are provided to detect a rod drop and is the backup to detection by rapid reduction of ex-core neutron flux. The dropping of a rod assembly can occur when the rod drive mechanism is de-energized from the Rod Control System. This accident has been evaluated in the UFSAR and in all cases the DNB design bases is met by demonstration that the DNBR is greater than the limiting value. Thus, this change deleting the monthly analog rod position test does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The digital-based IRPI system continues to meet the design function of providing reliable control rod position indication. The proposed change and associated replacements with digital-based IRPI system electronics provides enhanced testing through the automatic self-testing diagnostic features. Consequently, the overall ability to detect failures is not degraded. Therefore, the change deleting the monthly analog rod position test does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendments request: December 19, 2002.

Description of amendments request: The proposed changes would revise the Technical Specifications (TS) to Facility Operating License Nos. DRP–32 and DRP–37 for Surry Power Station, Units 1 and 2, respectively, to reflect changes in regulations, correct typographical and editorial errors made in previous TS revisions, and to revise TS cross-references to Updated Final Safety Analysis Report sections.

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:

Dominion has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed administrative change to the Surry Power Station Units 1 and 2 Technical Specifications (TS) and Bases. The proposed change to the Surry TS makes administrative revisions to reflect changes in regulations, corrects editorial and typographical errors from previous TS revisions, and revises TS cross-references to Updated Final Safety Analysis Report (UFSAR) sections. Due to the strictly administrative nature of the proposed TS change, we have determined that a significant hazards consideration does not exist. The basis for this determination is provided as follows:

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

The proposed change is administrative in nature and as such does not impact the condition or performance of any plant structure, system or component. The proposed administrative change does not affect the initiators of any previously analyzed event nor the assumed mitigation of accident or transient events. As a result, the proposed change to the Surry Technical Specifications does not involve any increase in the probability [nor] the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities or consequences are being affected by this proposed administrative change.

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

The proposed change is administrative in nature, and therefore does not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions and no new failure modes are being introduced. Therefore, the proposed administrative change to the Surry Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

The proposed change is administrative in nature, and does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the administrative ''cleanup'' of the Surry Technical Specifications. Therefore, the proposed administrative change to the Surry Technical Specifications does not involve a reduction in a margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov.*

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

Date of application for amendment: December 19, 2001, as supplemented July 30, 2002, and November 14, 2002.

Brief description of amendment: The amendment includes a revision of the Technical Specification (TS) Limiting Conditions for Operation 3.4, ''Decay Heat Removal Capability,'' conforming changes to TS Table 3.5–2, ''Accident Monitoring Instruments,'' and TS 4.9.1.2, ''Decay Heat Removal—Periodic Testing,'' and numerous editorial changes.

Date of issuance: January 16, 2003.

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

Amendment No.: 242.

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

Date of initial notice in ***Federal Register:*** March 19, 2002 (67 FR 12598).

The supplements dated July 30, and November 14, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated January 16, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 2, 2002.

Brief description of amendments: These amendments change the administrative controls in Technical Specification 5.7, ''High Radiation Area.''

Date of issuance: January 13, 2003.

Effective date: January 13, 2003.

Amendment Nos.: 225 and 252.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50950).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 13, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 14, 2002, as supplemented by letter dated December 17, 2002.

Brief description of amendment: The amendment revised Technical Specification Table 3.3.8.1–1, ''Loss of Power Instrumentation,'' by changing the degraded voltage—voltage basis and loss-of-coolant accident time delay allowable values to reflect the results of new calculations performed in association with a design basis reconstitution.

Date of issuance: January 16, 2003.

Effective date: As of the date of issuance and shall be implemented no later than November 30, 2003.

Amendment No.: 128.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42823).

The December 17, 2002, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2003.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington Date of application for amendment: October 22, 2002.

Brief description of amendment: The amendment revises the Technical Specifications (TS) to change TS section 5.0, ''Administrative Controls,'' and adopt Technical Specification Task Force (TSTF) -258, Revision 4. The change revises: (1) Section 5.2.2, ''Unit Staff,'' to delete details of staffing requirements and delete requirements for the Shift Technical Advisor (STA) as a separate position while retaining the function, (2) section 5.5.4, ''Radioactive Effluent Controls Program,'' to be consistent with the intent of 10 CFR part 20, (3) section 5.6.4, ''Monthly Operating Reports,'' to delete periodic reporting requirements for main steam safety/relief valve challenges to be consistent with Generic Letter 97–02, ''Revised Contents of the Monthly Operating Report,'' and (4) section 5.7, ''High Radiation Area,'' in accordance with 10 CFR 20.1601(c). TS section 5.3.2 is added to incorporate regulatory definitions for the senior reactor operator (SRO) and reactor operator (RO) positions.

Date of issuance: January 9, 2003.

Effective date: January 9, 2003, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 182.

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

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75870).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 28, 2002.

Brief description of amendment: The amendment revised Technical Specification (TS) sections 3.7, ''Auxiliary Electrical Systems,'' and 4.6, ''Emergency Power System Periodic Tests,'' to relocate the requirements for the gas turbine generators to the Updated Final Safety Analysis Report (UFSAR) and the plans, programs and procedures that document and control the credited functions of these systems, structures, and components. The amendments also deleted TS 3.7.B.2.b. to remove the option that allows power operation for up to 72 hours with a gas turbine as the only available 13.8 kilovolt power source.

Date of issuance: January 17, 2003.

Effective date: This license amendment is effective as of the date of its issuance and shall be implemented within 60 days and only after incorporation of the required changes into the UFSAR and completion of the necessary implementation and procedural changes.

Amendment No.: 236.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications and Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34484).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 5, 2002.

Brief description of amendment: The amendment relocates Technical Specification (TS) 3/4.6.I to the Pilgrim Nuclear Power Station Updated Final Safety Analysis Report. The affected TS contains snubber operability and surveillance requirements. The associated Bases section will also be relocated.

Date of issuance: January 14, 2003.

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

Amendment No.: 195.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68735).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 27, 2002.

Brief description of amendments: The amendments revise Appendix B, ''Environmental Protection Plan (Non-Radiological),'' of the licenses to remove a parenthetical reference to a superseded section of 10 CFR part 51.

Date of issuance: January 21, 2003.

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

Amendment Nos.: 211 & 205.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised Appendix B of the licenses.

Date of initial notice in ***Federal Register:*** October 29, 2002 (67 FR 66009).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: October 8, 2002.

Brief description of amendment: The amendment relocates Technical Specification 2.13, ''Nuclear Detector Cooling System,'' and its associated Bases to the Fort Calhoun Station Updated Safety Analysis Report.

Date of issuance: January 16, 2003.

Effective date: January 16, 2003, and shall be implemented within 120 days of the date of issuance. Implementation includes the incorporation of changes to the Fort Calhoun Station Updated Safety Analysis Report as described in the licensee's application dated October 8, 2002.

Amendment No.: 214.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68741).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 22, 2002, as supplemented by letter dated October 8, 2002.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.19, Surveillance Requirement (SR) 3.0.4 and adds TS 5.20 and SR 3.0.5 to extend the delay period before entering a limiting condition for operation following a missed surveillance.

Date of issuance: January 16, 2003.

Effective date: January 16, 2003, and shall be implemented with 120 days from the date of issuance, including the incorporation of changes to the technical specification Bases as described in the licensee's application dated July 22, 2002, as supplemented by letter dated October 8, 2002.

Amendment No.: 215.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 3, 2002 (67 FR 56326).

The supplemental letter of October 8, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2003.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: September 23, 2002.

Brief description of amendments: These amendments revised Technical Specification (TS) section 2.6.2.4, ''Residual Heat Removal [RHR] Suppression Pool Cooling,'' to adopt TS Task Force (TF) change 230, Revision 1 (TSTF–230, Revision 1). This change to Required Action B of Limiting Condition for Operation 3.6.2.3 allows two RHR suppression pool cooling subsystems to be inoperable for up to 8 hours.

Date of issuance: January 16, 2003.

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

Amendment Nos.: 207, 181.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 29, 2002 (67 FR 66012). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 16, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 17, 2002, as supplemented December 31, 2002.

Brief description of amendment: The amendment provides a one-time change to Technical Specification (TS) 4.8.1.1.2.h.14 to allow the testing of Hope Creek's emergency diesel generator (EDG) lockout relays to be performed at power until startup from its eleventh refueling outage (spring 2003). The current TS surveillance requirement only allows the EDG lockout relays to be tested during shutdown conditions. PSEG requested that the TS change be issued on an exigent basis in accordance with title 10 of the Code of Federal Regulations (10 CFR), part 50, section 50.91(a)(6). Approval and implementation of the TS change allows the testing that has been completed to be used to comply with TS 4.8.1.1.2.h.14.

Date of issuance: January 10, 2003.

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

Amendment No.: 141.

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

Public Comments Requested as to Proposed No Significant Hazards Consideration: Yes (67 FR 79163) December 27, 2002. That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by January 27, 2003, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after the issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, final determination of no significant hazards consideration, and state consultation are contained in a safety evaluation dated January 10, 2003.

NRC Section Chief: James W. Clifford.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendments request: February 18, 2002, as supplemented in letter dated July 23, 2002.

Brief description of amendments: The proposed amendments revised

Technical Specification 3/4.6.1.7, ''Containment Ventilation System,'' to extend the intervals between operability tests of the normal and supplementary containment purge valves.

Date of issuance: January 7, 2003.

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

Amendment Nos.: Unit 1–147; Unit 2–135.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 19, 2002 (67 FR 12608). The July 23, 2002, supplemental letter provided clarifying information that was within the scope of the original **Federal Register** notice (67 FR 12608) and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 7, 2003.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project. Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 22, 2002.

Brief description of amendments: The proposed amendments revised Technical Specification 3/4.3.5, allowing the automatic operation of the atmospheric steam relief valves during Mode 2 to maintain secondary side pressure at or below an indicated steam generator pressure of 1225 psig during startup and shutdown of the reactors.

Date of issuance: January 13, 2003.

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

Amendment Nos.: Unit 1–148; Unit 2–136.

Facility Operating License Nos. NPF–76 and NPF–80: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 9, 2002 (67 FR 45571).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 13, 2003.

No significant hazards consideration comments received: No.

Dated in Rockville, Maryland, this 28th day of January 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–2415 Filed 2–3–03; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Excepted Service

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: This gives notice of OPM decisions, granting authority to make appointments under Schedule C in the excepted service as required by 5 CFR 6.1 and 213.103.

FOR FURTHER INFORMATION CONTACT: Pam Shivery, Director, Washington Service Center, Employment Service (202) 606–1015.

SUPPLEMENTARY INFORMATION: Appearing in the listing below are one Schedule A authority and the individual authorities established under Schedule C between between December 01, 2002 and December 31, 2002. Future notices will be published on the fourth Tuesday of each month, or as soon as possible thereafter. A consolidated listing of all authorities as of June 30 is published each year.

Schedule A

U.S. Chemical and Hazard Investigation Board

Up to 37 positions established to create the Chemical Safety Hazard Investigation Board. No new appointments may be made under this authority after December 31, 2000. Effective December 31, 2002.

Schedule C

Department of Agriculture

Special Assistant to the Administrator, Farm Service Agency. Effective December 10, 2002.

Special Assistant to the Chief of Staff. Effective December 12, 2002.

Assistant to the Chief for Environment and Natural Resources. Effective December 20, 2002.

Director of Marketing and Public Relations to the Chief Natural Resource and Conservation Service. Effective December 20, 2002.

Confidential Assistant to the Assistant Secretary for Congressional Relations. Effective December 20, 2002.

Department of Commerce

Senior Policy Advisor to the Assistant to the Secretary and Director, Office of Policy and Strategic Planning. Effective December 9, 2002.

Special Assistant to the Director of External Affairs. Effective December 30, 2002.

Special Assistant to the Director of External Affairs. Effective December 30, 2002.

Special Assistant to the Director of External Affairs. Effective December 30, 2002.

Department of Defense

Staff Assistant to the Deputy Assistant Secretary of Defense (Eurasia). Effective December 4, 2002.

Protocol Officer to the Director of Protocol. Effective December 9, 2002.

Confidential Assistant to the Assistant Secretary of Defense Command, Control, Communications and Intelligence. Effective December 30, 2002.

Special Assistant to the Assistant Secretary of Defense (Legislative Affairs). Effective December 30, 2002.

Special Assistant to the Assistant Secretary of Defense (Legislative Affairs). Effective December 30, 2002.

Defense Short to the Special Assistant to the Secretary of Defense for White House Liaison. Effective December 31, 2002.

Department of Education

Special Assistant to the Assistant Secretary for Management. Effective December 19, 2002.

Deputy Secretary's Regional Representative, Region IX to the Deputy Assistant Secretary for Regional Services. Effective December 20, 2002.

Confidential Assistant to the Special Assistant. Effective December 31, 2002.

Department of Energy

Special Assistant for Communications to the Director, Office of Civilian Radioactive Waste Management. Effective December 4, 2002.

Deputy Chief of Staff to the Deputy Secretary of Energy. Effective December 13, 2002.

Department of Health and Human Services

Special Assistant to the Assistant Secretary for Public Affairs. Effective December 2, 2002.

Director of Speechwriting to the Deputy Assistant Secretary for Public Affairs (Media). Effective December 10, 2002.

Assistant to the Commissioner for Presidential Initiatives. Effective December 12, 2002.