Agency Action Review Meeting (Public Meeting) (Contact: Robert Pascarelli, 301–415–1245). Afternoon session.

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

Week of May 19, 2003—Tentative

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

Week of May 26, 2003—Tentative

Wednesday, May 28, 2003

9:30 a.m. Meeting with Advisory Committee on the Medical Uses of Isotopes (ACMUI) (Public Meeting) (Contact: Angela Williamson, 301–415–5030)

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

2:45 p.m. Discussion of Management Issues (Closed—Ex. 2)

Thursday, May 29, 2003

9:30 a.m. Briefing on Status of Revisions to the Regulatory Framework for Steam Generator Tube Integrity (Public Meeting) (Contact: Louise Lund, 301–415–3248)

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

2 p.m. Briefing on Equal Employment Opportunity Program (Public Meeting) (Contact: Corenthis Kelley, 301–415–7380)

Week of June 2, 2003—Tentative

There are no meetings scheduled for the Week of June 2, 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.

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

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

Dated: April 24, 2003.

D.L. Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–10608 Filed 4–25–03; 10:33 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 18, 2003, through May 1, 2003. The last biweekly notice was published on April 15, 2003, (68 FR 18269).

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

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

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

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

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

By May 29, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

Non-timely 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.*

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

Date of amendment request: March 20, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) and the licensing basis in the Updated Safety Analysis Report (UFSAR) to support installation of a passive low-pressure injection (LPI) cross connect inside containment. The proposed changes to the TS would add requirements for the passive LPI cross connect and eliminate requirements associated with the capability to cross connect by manual operator action the trains outside containment. The proposed changes to the UFSAR would revise the licensing basis for a portion of the core flood and LPI/Decay Heat Removal (DHR) piping to allow the exclusion of dynamic effects associated with postulated pipe rupture of that piping by application of leak-before-break technology for Unit 1. The proposed changes to the UFSAR would also revise the licensing basis for selected portions of the LPI/DHR piping to adopt the design requirements of Standard Review Plan Section 3.6.2, Branch Technical Position MEB 3–1.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures 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: The proposed LAR [Licence Amendment Request] modifies the Technical Specifications [(TS)] to incorporate new TS requirements associated with the new Low Pressure Injection (LPI) System configuration and eliminate TS requirements associated with the old LPI configuration. The proposed LAR also modifies the licensing basis to adopt Standard Review Plan (SRP) 3.6.2 Branch Technical Position (BTP) MEB 3–1 requirements for selected portions of LPI piping and to credit Leak-Before-Break (LBB) to allow the dynamic effects associated with postulated pipe rupture of selected portions of the LPI/Core Flood (CF) piping to be excluded from the design basis. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet GDC [General Design Criterion] 4 requirements related to environmental and dynamic effects. The proposed LAR will continue to ensure that ONS [Oconee Nuclear Station] can meet design basis requirements associated with the LPI safety function. The LPI System provides a means for delivering a large volume of borated water to the reactor core following postulated large pipe breaks in the Reactor Coolant System. The planned modification adds a passive crossover connection between the two LPI injection lines inside containment, along with necessary check valves and flow orifices that will eliminate the need for time-critical operator actions to manually open the LPI discharge header outside containment. The new components will have the same pressure, seismic, and quality group qualifications as the existing components in the LPI system. The addition of the crossover line will enhance the ability of the control room operator to mitigate the consequences of specific events for which LPI is credited. Therefore, the proposed LAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The LPI system is also relied on to cool the reactor core during unit shutdown. Hydraulic analyses have demonstrated that adequate LPI flow is available for normal shutdown cooling with the new LPI piping configuration.

2. Create the possibility of a new or different kind of accident from any kind of accident previously evaluated: The proposed LAR modifies the Technical Specification to incorporate new TS requirements associated with the new LPI System configuration and eliminate TS requirements associated with the old LPI System configuration. The proposed LAR also modifies the licensing basis to adopt MEB 3–1 requirement for selected portions of LPI piping and to credit LBB to allow the dynamic effects associated with postulated pipe rupture of selected portions of the LPI/Core Flood (CF) piping to be excluded from the design basis. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet GDC 4 requirements related to environmental and dynamic effects. The LPI and Core Flood systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS and licensing basis changes do not affect the mitigating function of these systems. Consequently, these changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment 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 TS and licensing basis changes do not unfavorably affect any plant safety limits, set points, or design parameters. The changes also do not unfavorably affect the fuel, fuel cladding, RCS, or containment integrity. Therefore, the proposed TS and licensing basis changes, which adds TS requirements and adopts new design allowances associated with the passive LPI cross connect modification, do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 19, 2003.

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

The changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated March 19, 2003.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below: Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont *Date of amendment request:* March 26, 2003.

Description of amendment request: The amendment request proposes to adopt the Boiling Water Reactor Vessel and Internals Project integrated surveillance program (BWRVIP ISP) as the basis for demonstrating compliance with the requirements of Appendix H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), ''Reactor Vessel Material Surveillance Program Requirements'' and delete Technical Specification (TS) 4.6.A.5. The licensee also proposes to update the pressure-temperature (P–T) curves through the end of the current operating license by revising TS Figures 3.6.1, 3.6.2, and 3.6.3.

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

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

Brittle fracture of the reactor pressure vessel (RPV) is not a postulated or evaluated design basis accident. No evaluations of other postulated accidents are affected by this proposed change. Because the applicable regulatory requirements continue to be met, the change does not significantly increase the probability of any accident previously evaluated.

Also, the change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

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 involve a modification of the design of plant structures, systems, or components. The change will not impact the manner in which the plant is operated and 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 affected, and no severe testing of equipment will be imposed. No new failure modes or mechanisms will be introduced as a result of this proposed change.

Therefore, the changes to the material surveillance program and pressure-temperature limits that compose 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 assumption 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 any increase in calculated off-site dose consequences.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: February 3, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) 3/4.7.1.4, ''Turbine Cycle—Specific Activity,'' and its associated bases. With the exception of TS 4.0.4, wording similar to that presented in the improved Standard Technical Specifications will be adopted. The amendment request proposes an exception to the requirements of TS 4.0.4 when entering MODE 4, along with conditions for when the surveillance requirement must be satisfied in MODE 4. Additionally there are editorial changes to the TS Index reflecting the proposed revision.

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

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

The proposed changes, in part, modify the modes of applicability by stating that TS 4.0.4 is not applicable for Mode 4 entry. For the surveillance requirement, the change specifies the conditions in Mode 4 that are necessary to obtain a representative sample from the steam generators. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The level of specific activity contained in the reactor coolant is germane to the consequences of an accident and 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 unanalyzed 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 a previously analyzed event 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 changes do not alter the initial conditions assumed in the analysis of interest. The plant parameters assumed for the analyses are maintained within assumed limits through compliance with the Technical Specifications and plant procedures. Additionally, the proposed changes do not impose any new safety analyses limits. Any deviation from the allowable activity limits will require the plant to be placed in a condition where the specification does not apply. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do 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 that 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. These changes have no physical effect on any plant equipment. Therefore, the changes do not create the possibility of a new of different kind of accident from any previously evaluated.

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

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 these changes, no operating parameters are revised, and no changes are proposed to the actuation setpoints. The limit on secondary coolant Dose Equivalent Iodine remains at the current value of 0.1 microcuries per gram. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit 1 (NMP1), Oswego County, New York

Date of amendment request: October 7, 2002, as supplemented on March 24, 2003.

Description of amendment request: The licensee's October 7, 2002, application proposed to add Specification 4.0.3 to address missed surveillances to the Technical Specifications (TSs). This new specification specifies an initial 24-hour delay period for performing a missed surveillance prescribed by Specification 3.0.3. Specification 4.0.3 will also require: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' In addition, the licensee proposed to add wording to each of the following existing specifications such that the new Specification 4.0.3 would apply to them: Specification 6.16, 6.17, 6.18, and 6.19. On November 12, 2002, the Nuclear Regulatory Commission (NRC) staff published a proposed no significant hazards consideration determination and opportunity for a hearing (67 FR 68739) for the October 7, 2002, application.

As a result of the NRC staff comments, the licensee supplemented the application by a letter dated March 24, 2003. The supplement adds new requirements related to the use and application of the surveillance requirements (SRs) currently included in the TSs.

These new explicit SR applicability requirements would supersede the more general current requirements. The proposed new requirements reflect the current practices at NMP1, and as such, do not change any existing method of plant operation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration for the March 24, 2003, supplement, 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.

Adoption of new administrative requirements related to the proper use of the surveillance requirements currently included in the NMP1 TSs do not affect any accident initiator, and as such, will have no effect on the probability of an accident. The proposed changes do not involve physical changes to the plant or introduce any new modes of operation. Accordingly, continued assurance is provided that the process variables, structures, systems, and components are maintained such that there will be no degradation of any fission product barrier which could increase the radiological consequences of an accident. Therefore, the proposed changes do 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.

Adoption of new administrative requirements related to the proper use of the surveillance requirements currently included in the NMP1 TSs will have no adverse effect on the design or assumed accident performance of any structure, system, or component, or introduce any new modes of system operation or failure modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed changes add new administrative requirements related to the proper use of the surveillance requirements currently included in the NMP1 TSs. The addition of requirements will make application of the surveillance requirements more restrictive than currently required by the TSs. Accordingly, 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 supplement of March 24, 2003, involves no significant hazards consideration.

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

NRC Section Chief: Richard J. Laufer.

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: March 27, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement 3.1.4.1, ''Rod Group Alignment Limits, to change the allowable alignment limits of individual rods in Mode 1 when greater than 85-percent power.

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.

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.

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.

Based on the analyses documented in WCAP–15432, Revision 2 [''Conditional Extension of the Rod Misalignment Technical Specification for Point Beach Units 1 and 2, (proprietary)'' dated April 2001], all pertinent licensing-basis acceptance criteria have been met and the margin of safety, as defined in the Technical Specification Bases, is not significantly reduced in any of the Point Beach licensing basis accident analyses due to the subject change. Therefore, the probability of an accident previously evaluated has not significantly increased. Because design limitations continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Neither rod position indication nor the limits on allowed rod position deviation is an accident initiator or precursor. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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.

The changes described in the proposed amendment are supported by the analyses provided in the submittal [the March 27, 2003, application]. The evaluation of the effects of the proposed changes indicates that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of any new or different accident nor create any new failure mechanisms.

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.

Based on the analyses documented in WCAP–15432, Revision 2, all pertinent licensing-basis acceptance criteria have been met and the margin of safety, as defined in the Technical Specification

Bases, is not significantly reduced in any of the Point Beach licensing basis accident analyses based on the subject changes to safety analyses input parameter values. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. Since the analyses in the accompanying submittals [March 27, 2003, application and WCAP–15432] demonstrate that all applicable acceptance criteria continue to be met, the subject operating conditions will not involve a significant reduction in a margin of safety at Point Beach.

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.

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

Date of amendment request: March 25, 2003.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.2.1 and TS 3.2.3 for implementation of relaxed axial offset control of the reactor cores, relocate selected operating parameters from TS 2.0 and TS 3.3.1 to the Core Operating Limits Report (COLR), revise the Pressurizer Pressure-Low Allowable Value, and revise the appropriate references in TS 5.6.5 to the NRC-approved methodologies which support relocation of operating parameters to the COLR.

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:

Group 1—Implementation of Relaxed Axial Offset Control

A. TS 3.2.1, Heat Flux Hot Channel Factor—$F_Q(Z)$ and Bases: Modification of Required Actions and Completion Time if $FW_Q(Z)$ is not within its limit and update Bases.

B. TS 3.2.3, Axial Flux Difference (AFD) and Bases: Modification of Limiting Conditions for Operation, Actions and Surveillance Requirements and revision of the Bases.

This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits.

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

This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits. The revised Technical Specifications and parameter changes associated with relaxed axial offset control assure that the limiting safety analysis inputs (such as, heat flux hot channel factor and axial flux difference limits) are not exceeded. The bounding power distribution transient factor values, W(Z), and the axial flux difference limits that are documented in the Core Operating Limits Report will be determined by NRC approved analytical methods and will be validated as part of the cycle specific reload evaluation process.

Heat flux hot channel factors and axial flux difference limits are not assumed accident initiators. Therefore, the relaxed axial offset control related Technical Specification changes do not involve a significant increase in the probability of an accident.

Likewise, operation of the plant within the proposed Technical Specification controls and limits assures that safety analysis assumptions are met, thus, if an accident were to occur, the consequences would continue to be bounded by the accident analyses. Therefore, the relaxed axial offset control related technical specification changes do not involve a significant increase in the consequences of an accident.

The relaxed axial offset control related technical specification changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

This license amendment request proposes to revise the Technical Specifications to implement the relaxed axial offset control methodology to address the heat flux hot channel factor and axial flux difference limits. The supporting Technical Specification limits are defined by NRC approved analytical methods which are performed to conservatively bound the operating conditions defined by the Technical Specifications and to demonstrate meeting the regulatory acceptance limits. The heat flux hot channel factor licensed safety margins are maintained. The heat flux hot channel factor conforms to plant design bases and limits actual plant operation within analyzed and licensed boundaries. The relaxed axial offset control methodology has been demonstrated to ensure that core heat flux hot channel factors will remain below accident analysis limits. The margin of safety provided by the analyses in accordance with the acceptance limits is maintained and not reduced. Thus, the implementation of relaxed axial offset control at Prairie Island does not involve a significant reduction in a margin of safety.

Group 2—Relocation of Technical Specifications Safety Limits Figure and Overtemperature Delta-T and Overpower Delta-T Parameter Values to the Core Operating Limits Report, and Miscellaneous Administrative Changes

A. TS 2.1.1, ''Reactor Core SLs [Safety Limits]'' and Bases: Relocate the safety limits Figure to the Core Operating Limits Report, update TS 2.1.1 and Bases.

B. TS 3.3.1, Table 3.3.1–1 (Pages 2, 7 and 8), ''Reactor Trip System Instrumentation'', Overpower Delta-T Trip Function, and Overtemperature Delta-T and Overpower Delta–T parameter values: Delete SR [Surveillance Requirement] 3.3.1.3, SR 3.3.1.6, and remove f(DI) from Overpower Delta-T Trip Function, relocate overtemperature delta-T and overpower delta-T parameter values and revise the Bases.

C. TS 5.6.5, Core Operating Limits Report (COLR): Additions to document Technical Specifications with limits in the Core Operating Limits Report and the analytical methods used to determine the values for relocated safety limits and overtemperature delta-T and overpower delta-T parameters and miscellaneous administrative changes.

This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report. Relocation of these limits and parameter values to the Core Operating Limits Report allows them to be changed under licensee controls. This license amendment also proposes to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which will be used to determine the safety limits and overtemperature delta-T and overpower delta-T parameter values. These changes are acceptable because the values used to operate the Prairie Island plant will be determined using NRC approved methods and these changes are consistent with the guidance of the industry standard Technical Specifications, NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plants''. This license amendment request also proposes to delete references to an NRC Safety Evaluation and make some editorial corrections in the Technical Specifications administrative controls section. These changes are acceptable since they are administrative and do not affect plant operation.

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

This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report and to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which support determination of these limits and parameter values. The safety limits and overtemperature delta-T and overpower delta-T parameter values that are documented in the Core Operating Limits Report will be determined by NRC approved analytical methods and will be validated as part of the cycle specific reload evaluation process.

Safety limits are not assumed accident initiators. Thus relocation of the safety limits does not involve a significant increase in the probability of an accident. Overtemperature delta-T and overpower delta-T parameter values are inputs to the reactor trip system which is provided to mitigate the consequences of an accident. The reactor trip system is not an accident initiator and therefore, changes to input values do not increase the probability of an accident.

Safety limits define bounding values within which plant operation will not initiate an accident condition. Safety limits relocated to the Core Operating Limits Report and determined by use of NRC approved methodologies will continue to determine the safe limits of plant operation, thus this change does not involve a significant increase in the consequences of an accident. The reactor trip system, with inputs from the overtemperature delta-T and overpower delta-T trip functions, mitigates the consequences of accidents.

The overtemperature delta-T and overpower delta-T trip parameter values are determined to assure that the design limit departure from nucleate boiling ratio is met and fuel integrity is maintained. Overtemperature delta-T and overpower delta-T trip parameters relocated to the Core Operating Limits Report and values determined by use of NRC approved methodologies will continue to determine the inputs for these trip functions which mitigate the design basis accident consequences, thus this change does not involve a significant increase in the consequences of an accident.

Addition of references to NRC approved methodologies in the Technical Specifications administrative controls section is an administrative change which does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed miscellaneous administrative changes in the Technical Specifications administrative controls section do not affect plant operation and therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be impacted as a result of the proposed technical specification changes. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created. The proposed administrative changes do not create the possibility of a new or different kind of accident from those previously analyzed.

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

This license amendment request proposes to relocate the safety limits and overtemperature delta-T and overpower delta-T parameter values to the Core Operating Limits Report and to include, in the Technical Specifications administrative controls section, the appropriate references to the NRC approved methodologies which support determination of these limits and parameter values. This proposed change also allows these relocated limits and parameter values to be changed under licensee controls. Safety limits in the Core Operating Limits Report will be determined by use of NRC approved methodologies and will continue to determine the safe limits of plant operation. Overtemperature delta-T and overpower delta-T trip parameter values in the Core Operating Limits Report will be determined by use of NRC approved methodologies and will continue to determine the inputs for these trip functions which mitigate design basis accidents. The Safety Limits licensed safety margins are maintained. The Safety Limits conform to plant design bases and limit actual plant operation within analyzed and licensed boundaries. The methodology described in WCAP–8745, along with the low pressurizer pressure allowable value, ensures that the overtemperature delta-T and overpower delta-T trips will protect against fuel centerline melting and departure from nucleate boiling during Condition II events. Thus, these changes do not involve a significant reduction in the margin of safety.

This license amendment request proposes to delete references to an NRC Safety Evaluation and make some editorial corrections in the Technical Specifications administrative controls section. These changes are administrative and thus do not involve a significant reduction in the margin of safety.

Group 3—Revision of Pressurizer Pressure-Low reactor trip Allowable Value

TS 3.3.1, Table 3.3.1–1 (Page 2), ''Reactor Trip System Instrumentation'', Function 8.a, Pressurizer Pressure-Low: Increase Pressurizer Pressure-Low Allowable Value.

This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1–1 for the Pressurizer Pressure-Low reactor trip.

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

This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1–1 for the Pressurizer Pressure-Low reactor trip. Pressurizer Pressure-Low reactor trip is an input to the reactor trip system which is provided to mitigate the consequences of an accident. The reactor trip system is not an accident initiator and therefore, changes to the Pressurizer Pressure-Low Allowable Value do not involve an increase in the probability of an accident.

The Pressurizer Pressure-Low Allowable Value is being increased which is a conservative change. The increase in the Pressurizer Pressure-Low reactor trip Allowable Value will assure that the overtemperature delta-T and overpower delta-T reactor trip functions, with values determined in accordance with NRC approved methodologies, provide protection against fuel centerline melting and departure from nucleate boiling for overpower and overtemperature events. Therefore, this change does not involve an increase in the consequences of an accident previously evaluated.

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

This proposed change does not involve a physical alteration of the plant; that is, no new or different type of equipment will be installed. This proposed change does not introduce any new mode of plant operation or change the methods governing normal plant operation. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

This license amendment request proposes to increase the Allowable Value defined in Table 3.3.1–1 for the Pressurizer Pressure-Low reactor trip. The Allowable Value is determined in accordance with an NRC accepted setpoint methodology with input from NRC approved analytical methods. These determinations are performed to conservatively bound the operating conditions defined by the Technical Specifications and to demonstrate meeting the regulatory acceptance limits.

Performance of analyses and evaluations for the cycle specific reload evaluation process will confirm that the operating envelope defined by the Technical Specifications continues to be bounded by the analytical basis and in no case exceeds the acceptance limits. The proposed Pressurizer Pressure-Low Allowable Value along with the overtemperature delta-T and overpower delta-T trips will protect against fuel centerline melting and departure from nucleate boiling during Condition II events. The proposed Allowable Value conforms to plant design bases and limits actual plant operation within analyzed and licensed boundaries. The margin of safety provided by the proposed Pressurizer Pressure-Low Allowable Value is maintained and not reduced. Thus, the increase in the Pressurizer Pressure-Low reactor trip Allowable Value does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 3, 2003.

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

The proposed changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post-Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated March 3, 2003.

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

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: December 23, 2002.

Description of amendment request: The amendment would change the Hope

Creek Generating Station (HCGS) reactor vessel material surveillance program required by Appendix H to Title 10 of the Code of Federal Regulations (10 CFR) part 50. This change would incorporate the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program (ISP) into the HCGS 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 change involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed change implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph Ill.C of Appendix H to 10 CFR 50. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated. The proposed change provides the same assurance of RPV [reactor pressure vessel] integrity. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No.

The proposed change revises the HCGS licensing basis to reflect participation in the ISP. The proposed change does not involve a modification of the design of plant structures, systems or components (SSC). Also, the proposed change will not degrade the reliability of SSCs important to safety since protective features will not be deleted or modified. The proposed change will not impact the manner in which the plant is normally operated. The proposed change maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Response: No.

The proposed change has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. Therefore, these changes do not involve a significant reduction in margin of safety.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, PO 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: September 20, 2002, as revised on February 14, 2003. This notice supercedes a previous notice (67 FR 75884) published on December 10, 2002, which was based on the licensee's application dated September 20, 2002.

Description of amendment request: The proposed amendment will: (1) Add a new limiting condition for operation (LCO) for spent fuel pool (SFP) boron concentration; (2) relocate requirements from Technical Specification (TS) Section 5.0, ''Design Features,'' to a new LCO in TS Section 3/4.7; and (3) revise existing TS 3/4.9.1 for refueling operations by relocating requirements for boron concentration to the Core Operating Limits Report (COLR) described in TS 6.9.1.9. The licensee also proposed related changes to the TS Bases. By letter dated February 14, 2003, PSEG revised its request, including lowering the minimum SFP boron concentration from 2300 parts per million (ppm) to 800 ppm.

Therefore, this notice supercedes a previous notice published on December 10, 2002, to reflect this change.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR), Section 50.91(a), the licensee has provided a revised analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The licensee proposed to change the Salem Nuclear Generating Station (Salem) TSs by: (1) adding a new LCO for SFP boron concentration; (2) relocating requirements from TS Section 5.0, ''Design Features,'' to a new LCO in TS Section 3/4.7; and (3) revising existing TS 3/4.9.1 for refueling operations by relocating requirements for boron concentration to the COLR. These changes are consistent with applicable LCOs in NUREG–1431, Revision 2, ''Improved Standard Technical Specifications, Westinghouse Plants,'' and will continue to provide administrative controls to ensure that a proper boron concentration is maintained in accordance with Salem's accident analyses. Because there are no changes to any of the input assumptions associated with postulated accidents involving refueling operations and the SFP, the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

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

Adding new LCOs for boron concentration in the SFP and relocating boron concentration requirements to the COLR will not change the conduct of operations in the SFP, refueling cavity and fuel transfer tube at Salem. Therefore, because plant operations will not change, the proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Refueling operations and SFP boron concentration limits will be based on approved methodologies and accident analyses that are unchanged as a result of the proposed TS amendments. Therefore, because existing margins of safety will be maintained, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment requests: February 28, 2003.

Description of amendment requests: The proposed license amendments would revise Action A of Technical Specification (TS) 3.5.2, ''ECCS—Operating,'' to change the completion time for restoring centrifugal charging pump (CCP) 1–1 to operable status during Diablo Canyon Power Plant (DCPP) Unit 1 Cycle 12, from 72 hours to 7 days. The 72-hour allowed completion time is not sufficient to accomplish such emergent repairs on an inoperable CCP. This license amendment request also removes a similar one-time change for DCPP Unit 2 CCP 2–1 which has expired.

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 emergency core cooling system (ECCS) and the centrifugal charging pumps (CCPs) are designed to respond to mitigate the consequences of an accident. They are not an accident initiator, and as such cannot increase the probability of an accident.

The loss of both CCPs, due to an inoperable CCP 1–1 and a single failure of CCP 1–2, could increase the consequences of an accident. A probabilistic risk assessment was performed to evaluate the increased consequences. The worst case risk increment due to the increased completion time for CCP 1–1 and the maximum allowed results in only a small quantitative impact on plant risk.

Allowing 7 days to complete the seal replacement and post-maintenance testing of CCP 1–1 is acceptable since the ECCS system remains capable of performing its intended function of providing at least the minimum flow assumed in the accident analyses. During the extended maintenance and test period, appropriate compensatory measures will be implemented to restrict high risk activity. The consequences of accidents, which rely on the ECCS system, will not be significantly affected.

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

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

There are no new failure modes or mechanisms created due to plant operation for an extended period to perform repairs and post-maintenance testing of CCP 1–1. Extended operation with an inoperable CCP does not involve any modification in the operational limits or physical design of the systems. There are no new accident precursors generated due to the extended allowed completion time.

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

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

Plant operation for seven days with an inoperable CCP 1–1 does not adversely affect the margin of safety. During the extended allowable completion time the ECCS system maintains the ability to perform its safety function of providing at least the minimum flow assumed in the accident analyses. During the extended maintenance and test period, appropriate compensatory measures will be implemented to restrict high-risk activity.

Therefore, the change does not involve a significant reduction in a margin of safety as defined in the basis for any Technical Specification.

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

Florida Power & Light Company, et al. (FPL's), Docket Nos. 50–335, and 50–389, St. Lucie Plant, Unit No. 1, and Unit No. 2, St. Lucie County, Florida

Date of amendment request: October 23, 2002.

Description of amendment request: The proposed license amendments would revise the Technical Specifications to include the design of a new cask pit spent fuel storage rack for each unit to increase the allowable spent fuel wet storage capacity at both units and include the description of Boral ™ as the neutron absorbing material used in the new cask pit storage racks. The proposal would also revise the spent fuel pool thermal-hydraulic analyses for core offload times and include a change in FPL's commitments regarding the Unit 2 spent fuel cooling system design basis described in the Updated Final Safety Analysis Report.

*Date of publication of individual notice in the **Federal Register:*** January 28, 2003 (68 FR 4244), as corrected March 31, 2003 (64 FR 15487).

Expiration date of individual notice: February 27, 2003.

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

Date of application for amendments: February 14, 2003.

Description of amendments request: Revise the Updated Final Analysis Report to change the methodology using a through-bolted connection frame that is different than the original design and construction of the steam generator roof compartment.

*Date of publication of individual notice in the **Federal Register:*** March 14, 2003 (68 FR 12382).

Expiration date of individual notice: April 14, 2003.

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

Date of application for amendments: March 18, 2003.

Description of amendments request: Revise the Updated Final Analysis Report to provide an alternative methodology using a Bar-Lock mechanical splice in lieu of the Cadweld splice used in the original design and construction of the concrete shield building dome.

*Date of publication of individual notice in the **Federal Register:*** March 17, 2003 (68 FR 12718).

Expiration date of individual notice: April 16, 2003.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: November 27, 2002.

Brief description of amendment: The amendment deleted Section 6.17, ''Post Accident Sampling,'' and thereby eliminating the requirements to have and maintain the subject system. The subject requirements were imposed by a July 7, 1981, Nuclear Regulatory Commission Confirmatory Order.

Date of Issuance: April 4, 2003.

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

Amendment No.: 237.

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

*Date of initial notice in **Federal Register:*** January 21, 2003 (68 FR 2798).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 4, 2003.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: June 11, 2002, as supplemented January 22, 2003.

Brief description of amendments: The amendment changes Technical Specifications 3.7.11 related to the operation of the spent fuel pool exhaust ventilation system during the movement of irradiated fuel assemblies.

Date of issuance: April 7, 2003.

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

Amendment Nos.: 234, 257.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 15, 2002 (67 FR 63689).

The January 22, 2003, supplemental letter provided clarifying information that did not enlarge the scope of the amendments as noticed in the original **Federal Register** notice or change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 7, 2003.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: July 17 and August 6, 2002.

Brief description of amendments: These amendments permit operation of Calvert Cliffs Unit 2 with a core containing up to eight lead fuel assemblies with fuel rods clad with an advanced zirconium-based alloy.

Date of issuance: April 14, 2003.

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

Amendment Nos.: 258 and 235.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 17, 2002 (67 FR 58637).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 14, 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: October 10, 2002, as supplemented on November 22, 2002, and January 28, 2003. The October 10, 2002, application replaced the original application dated December 12, 2001.

Brief description of amendment: This amendment changes Technical Specification (TS) Tables 3.2.A, 3.2.B, 4.2.A, and 4.2.B. The proposed changes affect various instrument trip level settings and decrease calibration frequencies for a variety of instruments. The proposed changes identify that the Reactor Water Cleanup (RWCU) system requires one channel in each of the two trip systems for each location. The proposed changes also clarify the titles of certain trip systems, move note numbers to their proper location, and correct a mis-referenced figure in a table note. Appropriate Bases pages were also changed to reflect the TS changes.

Date of issuance: April 17, 2003.

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

Amendment No.: 198.

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

*Date of initial notice in **Federal Register:*** February 18, 2003, (68 FR 7815).

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle 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: April 4, 2003.

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

Amendment Nos.: 132/132.

Facility Operating License Nos. NPF–37 and NPF–66: The amendments revised the Technical Specifications.

*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 April 4, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 4, 2002, as supplemented by letter dated February 19, 2003.

Brief description of amendment: This amendment revises Technical Specification (TS) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the amendment adds requirements to SR 4.0.3 to perform a risk evaluation for any Surveillance delayed greater than 24 hours and manage the risk impact, and specifies actions to be taken when a delayed surveillance is not performed or not met. The amendment is consistent with TS Task Force traveler TSTF–358, which has been approved by the Nuclear Regulatory Commission for incorporation into standard technical specifications in NUREG–1430. The TS Bases will be revised under the licensee's existing TS Bases control program to be consistent with the bases for TSTF–358.

Date of issuance: April 11, 2003.

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

Amendment No.: 254.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 7, 2003 (68 FR 804).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 10, 2002.

Brief description of amendment: This amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: April 17, 2003.

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

Amendment No.: 125.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 18, 2003 (68 FR 12954).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 14, 2002, as supplemented by letter dated January 20, 2003.

Brief description of amendment: This amendment revised technical specification (TS) 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time exception to Nuclear Energy Institute 94–01, ''Industry Guidance for Implementing Performance-Based Option of 10 CFR part 50 Appendix J,'' that extends the test interval of the containment integrated leak rate test from 10 to 15 years.

Date of issuance: April 8, 2003.

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

Amendment No.: 126.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 4, 2003 (68 FR 5676).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 8, 2003.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: August 15, 2002, as supplemented December 13, 2002.

Brief description of amendments: The amendments revise Technical Specifications Section 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 containment integrated leak rate test (ILRT). The changes were submitted on a risk-informed basis as described in Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis. The risk-informed analysis supporting the changes indicates that the increase in risk from extending the ILRT test interval from 10 to 15 years is insignificant.

Date of Issuance: April 10, 2003.

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

Amendment Nos.: 187 & 130.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 17, 2002 (67 FR 58647).

The supplement dated December 13, 2002, provided clarifying information that did not change the scope of the August 15, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 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: June 28, 2002, as supplemented December 18, 2002, January 18, 2003, and February 25, 2003.

Brief description of amendment: The amendment relaxes certain Technical Specifications (TSs) requirements for containment isolation and removes references to the Filtration Recirculation and Ventilation System charcoal filters.

Date of issuance: April 15, 2003.

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

Amendment No.: 146.

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

Date of initial notice in ***Federal Register:*** February 18, 2003 (68 FR 7818).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 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: October 9, 2002, as supplemented November 22, 2002, and December 6, 2002.

Brief description of amendment: The amendment grants, on a one-time basis, an extension of the Type A Integrated Leak Rate Test interval from 10 years to 15 years.

Date of issuance: April 16, 2003.

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

Amendment No.: 147.

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

Date of initial notice in ***Federal Register***: February 18, 2003 (68 FR 7819).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 23, 2002.

Brief description of amendments: The amendments revise the Salem Nuclear Generating Station, 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.

Date of issuance: April 10, 2003.

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

Amendment Nos.: 255 and 236.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: February 4, 2002 (68 FR 5681).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 2003.

No significant hazards consideration comments received: No.

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

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

Brief description of amendments: The amendments revise Technical Specifications (TSs) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of up to 24 hours to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the following requirement is added to SR 4.0.3: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' The amendments also add a requirement for a TS Bases Control Program to the administrative controls section of TSs and makes administrative changes to SRs 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plants.''

Date of issuance: April 16, 2003.

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

Amendment Nos.: 256 and 237.

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

Date of initial notice in ***Federal Register***: February 18, 2003 (68 FR 7820).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 1, 2002.

Description of amendments request: The amendments revised the Updated Safety Analysis Report (UFSAR) to eliminate consideration of a pressure regulator downscale failure as an abnormal operational transient.

Date of issuance: April 4, 2003.

Effective date: As of the date of issuance, to be incorporated into the UFSAR at the time of its next update.

Amendment Nos.: 244, 281 and 239.

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

Date of initial notice in ***Federal Register***: October 15, 2002 (67 FR 63697).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 2003.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of no Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a Federal Register notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess 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 Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 16, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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: Rulemakings 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 the 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 petition for leave to intervene and request for hearing 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of amendment request: April 14, 2003, as supplemented by letter dated April 15, 2003.

Description of amendment request: The amendment revises Limiting Condition for Operation (LCO) 3.7.5, ‘‘Control Building Chiller (CBC) System,’’ Required Action A.1 to add a provision that temporarily removes the restrictions of LCO 3.0.4 until May 16, 2003. This amendment allows entry into LCO 3.7.5 with an inoperable CBC subsystem.

Date of issuance: April 16, 2003.

Effective date: As of the date of issuance and shall be implemented immediately.

Amendment No.: 250.

Facility Operating License No. DPR–49: Amendment revises the technical specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated April 16, 2003.

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

NRC Section Chief: L. Raghavan.

Dated at Rockville, Maryland, this 21st day of April, 2003.

For the Nuclear Regulatory Commission

John A. Zwolinski,

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

[FR Doc. 03–10396 Filed 4–28–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

State of Wisconsin: NRC Staff Draft Assessment of a Proposed Agreement Between the Nuclear Regulatory Commission and the State of Wisconsin

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of a proposed agreement with the state of Wisconsin.

SUMMARY: By letter dated August 21, 2002, former Governor Scott McCallum of Wisconsin requested that the U. S. Nuclear Regulatory Commission (NRC) enter into an Agreement with the State as authorized by Section 274 of the Atomic Energy Act of 1954, as amended (Act).

Under the proposed Agreement, the Commission would relinquish, and Wisconsin would assume, portions of the Commission's regulatory authority exercised within the State. As required by the Act, NRC is publishing the proposed Agreement for public comment. NRC is also publishing the summary of a draft assessment by the NRC staff of the Wisconsin regulatory program. Comments are requested on the proposed Agreement and the staff's draft assessment which finds the Program adequate to protect public health and safety and compatible with NRC's program for regulation of Agreement material.

The proposed Agreement would release (exempt) persons who possess or use certain radioactive materials in Wisconsin from portions of the Commission's regulatory authority. The Act requires that NRC publish those exemptions. Notice is hereby given that the pertinent exemptions have been previously published in the **Federal Register** and are codified in the Commission's regulations as 10 CFR part 150.

DATES: The comment period expires May 8, 2003. Comments received after this date will be considered if it is practical to do so, but the Commission cannot assure consideration of comments received after the expiration date.

ADDRESSES: Written comments may be submitted to Mr. Michael T. Lesar, Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Washington, DC 20555–0001. Comments may be submitted electronically at *nrcrep@nrc.gov*.

The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/NRC/ADAMS/index.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*.

Copies of comments received by NRC may be examined at the NRC Public Document Room, 11555 Rockville Pike, Public File Area O–1–F21, Rockville, Maryland. Copies of the request for an Agreement by the Governor of Wisconsin including all information and documentation submitted in support of the request, and copies of the full text of the NRC Staff Draft Assessment are also available for public inspection in the NRC's Public Document Room—ADAMS Accession Numbers: ML030160104 and ML030900662.

FOR FURTHER INFORMATION CONTACT: Lloyd A. Bolling, Office of State and Tribal Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Telephone (301) 415–2327 or e-mail *LAB@nrc.gov*.

SUPPLEMENTARY INFORMATION: Since section 274 of the Act was added in 1959, the Commission has entered into Agreements with 32 States. The Agreement States currently regulate approximately 16,250 agreement material licenses, while NRC regulates approximately 4,900 licenses. Under the proposed Agreement, approximately 260 NRC licenses will transfer to Wisconsin. NRC periodically reviews the performance of the Agreement States to assure compliance with the provisions of section 274.

Section 274e requires that the terms of the proposed Agreement be published in the **Federal Register** for public