Summer Nuclear Station expires on August 6, 2022. The Virgil C. Summer Nuclear Station is a pressurized water reactor designed by Westinghouse Electric Corporation and is located in Fairfield County, South Carolina. The acceptability of the tendered application for docketing and other matters, including an opportunity to request a hearing, will be the subject of a subsequent **Federal Register** notice.

Copies of the application are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, or electronically from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). The ADAMS Public Electronic Reading Room is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html.* In addition, the application is available on the NRC web page at *http://www.nrc.gov/reactors/operating/licensing/renewal/applications.html,* while the application is under review. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC's PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

The license renewal application for the Virgil C. Summer Nuclear Station is also available to local residents at the Fairfield County Library, in Winnsboro, South Carolina, and at the Thomas Cooper Library, at the University of South Carolina in Columbia, South Carolina.

Dated at Rockville, Maryland, this 26th day of August, 2002.

For the Nuclear Regulatory Commission.

Pao-Tsin Kuo,

Program Director, License Renewal and Environmental Impacts Program, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 02–22331 Filed 8–30–02; 8:45 am]

BILLING CODE 7590–01–P

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, August 9, 2002, through August 22, 2002. The last biweekly notice was published on August 20, 2002 (67 FR 53983).

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

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

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

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

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

By October 3, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714,[1] which is available at the Commission's

[1] The most recent version of Title 10 of the CODE OF FEDERAL REGULATIONS, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: ''In all other circumstances, such ruling body or officer shall, in ruling on—

(1) A petition for leave to intervene or a request for hearing, consider the following factors, among other things:

(i) The nature of the petitioner's right under the Act to be made a party to the proceeding.

(ii) The nature and extent of the petitioner's property, financial, or other interest in the proceeding.

(iii) The possible effect of any order that may be entered in the proceeding on the petitioner's interest.

(2) The admissibility of a contention, refuse to admit a contention if:

(i) The contention and supporting material fail to satisfy the requirements of paragraph (b)(2) of this section; or

(ii) The contention, if proven, would be of no consequence in the proceeding because it would not entitle petitioner to relief.''

PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner 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 requested involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the rquest for a hearing. Any hearing held would take place after issuance of the amendment.

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to pdr@nrc.gov.

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 amendments request: June 11, 2002.

Description of amendments request: The proposed amendments revise the Unit 1 and 2 Technical Specification (TS) Administrative Controls Section to incorporate seven changes previously approved for the Improved Standard Technical Specifications (ISTS). These changes are reflected in Revision 2 of NUREG–1432 (Reference a). In addition, a change is also being requested to correct an inconsistency introduced in a prior TS amendment.

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

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

The majority of changes proposed are editorial in nature, that is, they do not change the fundamental requirement of the Technical Specification. They generally clarify the existing requirement. The remaining changes are changes to the Technical Specification requirements. The deletion of the pressurizer safety and relief valve challenges and failures report does not impact the operation of the pressurizer safety and relief valves and still permits reporting of significant failures under the provisions of 10 CFR 50.72 and 50.73. Removal of pipe supports from the Inservice Testing Program description corrects the description of the program. It does not change the manner or timing of any evaluations of pipe supports or snubbers. Removal of the discussion of the Nuclear Regulatory Commission environmental monitoring program with the state reflects the cancellation of that program with the state. It does not alter any other environmental monitoring requirements.

As described above, these proposed changes are generally editorial in nature or have no impact on plant operation. None of the proposed changes impact the operation of any equipment needed for the mitigation of an accident or any known accident initiators.

Therefore, the probability or consequences of an accident previously evaluated have not significantly increased.

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

As noted above, these changes are generally editorial in nature. That is, they do not change the fundamental requirement of the Technical Specification. They generally clarify the existing requirement. The remaining changes do not impact plant operation. None of the proposed changes would result in new or different plant operation or the addition of new equipment.

Therefore, the possibility of a new or different [kind] of accident from any previously evaluated is not created.

3. Would not involve a significant reduction in a margin of safety.

Since the majority of the proposed changes are editorial in nature, they do not change the fundamental Technical Specification requirement. Therefore, they do not impact the margin of safety represented by these Technical Specifications. The remaining changes do not impact plant operation and generally align these Technical Specification requirements with the criteria given in 10 CFR 50.36(c)(2)(ii). The deletion of the pressurizer safety and the relief valve challenges and failures report does not impact the operation of the pressurizer safety and relief valves and still permits reporting of significant failures under the provision of 10 CFR 50.72 and 50.73. Removal of pipe supports from the Inservice Testing Program description corrects the description of the program. It does not change the manner or timing of any evaluations of pipe supports or snubbers. Removal of the discussion of the Nuclear Regulatory Commission environmental monitoring program with the state reflects the cancellation of that program with the state. It does not alter any other environmental monitoring requirements. These changes do not impact the margin of safety.

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 amendments request involves no significant hazards consideration.

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

NRC Section Chief: Richard J. Laufer.

Carolina Power & Light Company (CP&L), Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of amendment request: May 10, 2002, as supplemented August 12, 2002.

Description of amendment request: The proposed amendment would allow an increase in the authorized reactor power level for HBRSEP2.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

The proposed change * * * does not involve a significant increase in the probability of an accident previously evaluated based on the results of comprehensive analytical efforts that were performed to demonstrate the acceptability of the proposed power uprate changes.

An evaluation has been performed that identified the systems and components that could be affected by these proposed changes. The evaluation determined that these systems and components will function as designed and that performance requirements remain acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pumps, steam generators and pressurizer) will continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components leading to an accident.

The Leak-Before-Break analysis conclusions remain valid and the breaks previously exempted from structural considerations remain unchanged.

Systems included within the scope of the Nuclear Steam Supply System (NSSS) will continue to perform their intended design functions during normal and accident conditions. Additionally, NSSS components will continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

The NSSS/Balance of Plant interface systems will continue to perform their intended design functions. The MSSVs [main steam safety valves] will provide adequate relief capacity to maintain the Main Steam System within design limits. The maximum feedwater flow rate and the isolation time for the MFRVs [main feedwater regulating valves] and Bypass Valves will continue to ensure that the analyzed containment pressure during postulated accidents remains below the allowable limit.

The current loss-of-coolant [accident] (LOCA) hydraulic analyses remain bounding.

The reduction in power measurement uncertainty achieved through the use of the Caldon Leading Edge Flow Meter (LEFM) Check-Plus™ system allows for certain safety analyses to continue to be used, without modification, at the 2346 MWt [megawatt thermal] power level (102 percent of 2300 MWt). Other safety analyses performed at a nominal power level of 2300 MWt have been either re-performed or re-evaluated to support the 2339 MWt power level, and continue to meet their applicable acceptance criteria. Some existing safety analyses had been previously performed at a power level greater than or equal to 2346 MWt, and thus continue to bound the 2339 MWt power level.

The proposed changes to the RCS [reactor coolant system] pressure-temperature limit curves impose a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50, Appendix G, ''Fracture Toughness Requirements,'' will continue to be met following the proposed power uprate. The design basis events that were protected against by these limits have not changed, therefore, the probability of an accident previously evaluated is not increased.

Based on the foregoing, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed power uprate changes. Systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component, and do not challenge the performance or integrity of any safety-related system.

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

Extensive analyses of the primary fission product barriers conducted in support of the proposed power uprate have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and compliance with regulatory acceptance criteria. As appropriate, evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC), or that are in compliance with applicable regulatory review guidance and standards.

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

Based on the above discussion, CP&L has determined that the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

Date of amendment request: August 8, 2002.

Description of amendment request: The proposed amendment allows a revision of the current reactor pressure vessel (RPV) material surveillance program description in the Updated Final Safety Analysis Report for Fermi 2 to reference the Integrated Surveillance Program (ISP) that was developed by the Boiling Water Reactor Owners Group's Vessel and Internals Project (BWRVIP). The proposed amendment is consistent with the NRC's Regulatory Issue Summary 2002–05, ''NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program,'' dated April 8, 2002 (ADAMS Accession No. ML020660522).

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

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

The proposed License Amendment involves a change in the program of RPV material surveillance for monitoring the effects of neutron embrittlement and thermal environment as required by Appendix H of 10 CFR 50. Instead of the Fermi 2 plant-specific program, the BWRVIP ISP is proposed for use in complying with the requirements of Appendix H [to 10 CFR Part 50]. Paragraph III.C of Appendix H provides the requirements for an ISP. The BWRVIP ISP has been reviewed and approved by the NRC staff as an acceptable program for use by all BWRs. There are many advantages for participating in the ISP over utilizing a plant-specific program. The advantages include improved compliance with the NRC requirements, better matching of the plant limiting material to the representative capsule material, additional data points for irradiated and unirradiated specimens, and better quality and consistency of the data and methodology. Additionally, future calculations of neutron fluence will be completed in accordance with the approved NRC methodologies in Regulatory Guide (RG) 1.190 [''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence''].

The data obtained from testing the RPV surveillance capsules is used to define the pressure-temperature limits for the RPV and to ensure that fracture toughness requirements for ferritic materials of pressure retaining components of the reactor coolant boundary are met. Using the ISP for RPV material surveillance program enhances the RPV integrity evaluations and results in using data from better-matching specimens. The ISP also results in better compliance with the NRC requirements and consistency among the BWR plants.

The proposed change results in better compliance with the regulatory requirements for RPV material surveillance; therefore, it does not increase the likelihood of a malfunction of plant structures, systems and components.

Based on the above, the proposed change does not significantly increase the probability or consequences of any accident previously evaluated.

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

The purpose of the RPV material surveillance program is to monitor neutron embrittlement and thermal environment effects in order to predict the behavioral characteristics of ferritic material of pressure retaining components of the reactor coolant pressure boundary and to ensure RPV fracture toughness and integrity requirements are not violated. The BWRVIP ISP was approved for use by all BWRs as an alternate to plant-specific programs. The change does not affect the design function or operation of any plant structure, system or component. The ISP is an approved alternate monitoring program that meets the regulatory requirements in Appendix H to 10 CFR 50. As an alternate monitoring program, the ISP cannot create a new failure mode involving the possibility of a new or different kind of accident. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The RPV material surveillance program requirements in Appendix H to 10 CFR 50 are designed to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the reactor coolant pressure boundary may be subjected over its service lifetime. The material surveillance data for the Fermi 2 RPV obtained through the ISP is equal or better to that from plant-specific programs. Paragraph III.C of Appendix H to 10 CFR 50 delineates the regulatory requirements for an ISP. The BWRVIP ISP meets these requirements and has been approved by the NRC. Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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: July 29, 2002.

Description of amendment request: The proposed amendments would revise Technical Specification Surveillance Requirement 3.7.2.2 to decrease the allowable closure time for the turbine stop valves from 15 seconds to 1 second.

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:

No. The request is for a decrease in the Turbine Stop Valve (TSV) closure time acceptance criteria of Technical Specification (TS) Surveillance Requirement (SR) 3.7.2.2, from a value of ≤15 seconds to a value of ≤1 second. This decrease in the closure time for the Channel B closure circuitry is more conservative and is being made to match the existing 1 second or less acceptance criteria of the closure time of the Channel A closure circuitry. The new Chapter 15 Transient Analysis Methodology assumes that the TSVs will be closed in 1 second or less by either the Channel A or Channel B closure circuitry. The new design has already been installed and tested, and is more conservative than the previous design. Therefore, the request for a more restrictive TS SR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

No. The 1 second or less closure time was and is acceptable under the existing TS SR for the Channel B circuitry since the existing acceptance is 15 seconds or less. This request is to change the TS SR and its Bases to a more restrictive requirement (1 second or less). This more restrictive requirement is being requested to ensure that the installed equipment will continue to meet the conditions and assumptions that are currently in the analysis model described in the Topical Report DPC–NE–3005–P, ''UFSAR Chapter 15 Transient Analysis Methodology''. Therefore, this request does not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

(3) Involve a significant reduction in a margin of safety:

No. The proposed change does not adversely affect any plant safety limits, setpoints, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. Therefore, the proposed change does not involve a reduction in a margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: July 18, 2002.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time 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 would be 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.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated July 18, 2002.

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

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

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

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

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

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

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

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

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: July 5, 2002.

Description of amendment request: The proposed amendment would increase the licensed power level by 1.5% from 1,998 megawatts thermal (MWt) to 2,028 MWt based on the installation of ultrasonic flow measurement instrumentation resulting in improved feedwater flow measurement accuracy. The proposed amendment would change the Operating License and Technical Specifications to reflect the increase in licensed power level.

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

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

The proposed increase in power level is achieved by improving the accuracy of the feedwater flow measurement instrumentation resulting in a more accurate feedwater flow used in the heat balance calculation. The increased flow accuracy improves the uncertainty in the core power level from the existing 2% margin to ≤0.5%. The probability of an accident previously evaluated is not increased by the proposed change because the flow measurement instrumentation is not an initiator of design-basis accidents (DBAs) evaluated in the updated final safety analysis report (UFSAR). The consequences due to postulated DBA events previously evaluated are based on analyses using a 2% margin above the current licensed power level which bounds the proposed 1.5% power level increase. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed increase in power level is achieved by improving the accuracy of the feedwater flow measurement instrumentation. Using the more accurate flow measurement in the heat balance calculation improves the core power level uncertainty. The proposed increase in power level will not create a change in the operation or function of the flow measurement instrumentation. Changes to the feedwater flow measurement accuracy does not create accident initiators not considered in the DBAs. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The calculated loads on all affected structures, systems, and components have been shown to remain within design criteria at the increased power level for all design-basis event categories. The current design margins, operational margins, and margins of safety are not exceeded by the increased power level. Therefore, the proposed 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: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: Jacob I. Zimmerman, Acting.

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

Date of amendment request: July 26, 2002.

Description of amendment request: The proposed amendment would allow relocation of the Kewaunee Nuclear Power Plant (KNPP) cycle dependent variables from the Technical Specifications (TS) to a formal report, Core Operating Limits Report (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:

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

The proposed changes relocate certain cycle specific parameters from the Technical Specifications to a Core Operating Limits Report (COLR) or are administrative in nature. Appropriate design and safety limits are retained or added to the Specifications thereby meeting the requirements of 10 CFR 50.36. Specific, approved methodologies used to determine and evaluate the parameter requirements are added to the Specifications and a reporting requirement is added to ensure the NRC is apprised of all changes. Approved methodologies are required to be used to evaluate and change parameters, and appropriate safety and design limits are maintained in the Technical Specifications. Thus, operation of KNPP will continue to meet all design and safety analysis requirements. Therefore, neither the probability nor consequences of an accident previously evaluated can be increased.

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

Operation of KNPP, in accordance with the proposed changes, will continue to meet all design and safety limits. Appropriate design and safety limits continue to be controlled within the Technical Specifications. These changes will not result in a change to the design and safety limits under which KNPP operation has been determined to be acceptable. Therefore, these changes cannot result in a new or different kind of accident from any accident previously evaluated.

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

Appropriate safety limits continue to be controlled by the Technical Specifications. Changes to cycle specific parameters related to these limits will be accomplished using NRC approved methodologies, thereby ensuring operation will continue within the bounds of the existing safety analyses including all applicable margins of safety. Therefore, operation in accordance with the proposed changes cannot result in a significant reduction in a margin of safety.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: July 26, 2002.

Description of amendment request: The proposed license amendment would implement changes to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) to accommodate Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features.

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

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

The [Nuclear Regulatory Commission] NRC generically approved Westinghouse 422V+

[Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features] fuel assemblies for use in reactors substantially similar to KNPP. NMC used 422V+ fuel in the Lead-Test-Assembly Program during cycle 25, as permitted by existing TS. Empirical data acquired during Cycle 25 confirms that this fuel is both compatible with KNPP reactor design and with the Framatome/ANP fuel currently in use. Reanalysis of postulated KNPP design basis accidents shows that reactor operation with 422V+ fuel remains within design basis limitations and safety margins. All design basis accidents and transients affected by the fuel upgrade were analyzed, and the results documented in the Westinghouse Report provided with this request. These analyses and evaluations show that use of 422V+ fuel is acceptable. The margin to safety is not exceeded in any instance. Pending approval of Addendum 2 to [Westinghouse Commercial Atomic Power ''Revision to Design Criteria''] WCAP 12488 revising the current transient stress strain criteria, all design basis acceptance criteria will be satisfied. Changes to the technical specification that remain within the limits of the bounding accident analyses cannot change the probability or consequence of an accident previously evaluated. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

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

Use of the 422V+ fuel is consistent with current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. The operational characteristics of 422V+ fuel are bounded by the safety analyses (Attachment 4 [of the submittal]). The 422V+ fuel design performs within existing fuel design limits. Thus, this proposal does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. Licensed safety margins are maintained. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Analyses of design basis accidents and transients were performed using power level greater than that currently licensed, thus rendering more conservative results than required. All safety analysis acceptance criteria are satisfied at this value and all KNPP safety requirements continue to be met. Use of 422V+ fuel as proposed by this amendment request is bounded by these analyses. Thus, changes proposed by this request 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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 22, 2002.

Description of amendment request: The proposed amendment would revise the reactor vessel pressure and temperature (P/T) limit curves in the Monticello Technical Specifications (TSs). The revised P/T limits will allow required hydrostatic and leak tests to be performed at a significantly lower temperature. This is expected to reduce challenges to plant operators associated with maintaining the reactor coolant system within a narrow temperature band during testing.

The Nuclear Management Company, LLC, is also requesting an exemption from the requirements of 10 CFR Part 50, Appendix G, to allow the use of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N–640 as the basis for these revised curves. The proposed P/T curves were developed in accordance with the 1989 edition of the ASME Code, Section XI, Appendix G; 10 CFR Part 50, Appendix G; and ASME Code Case N–640. The use of this Code Case as the basis for the proposed P/T curves constitutes an alternative to the requirements of 10 CFR Part 50, Appendix G. The regulation at 10 CFR 50.60(b) provides that the NRC may grant alternatives to the requirements in Appendix G by using the procedures for exemption specified in 10 CFR 50.12.

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

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

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the ASME Code and 10 CFR 50 Appendix G and H as restrictions on operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary.

The changes to the calculation methodology for the P/T limits are based upon ASME Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, division 1,'' and provide adequate margin in the prevention of a non-ductile type fracture of the reactor pressure vessel (RPV). The code case was developed based upon the knowledge gained through years of industry experience. The P/T limits developed using the allowances of ASME Code Case N–640 provide more operating margin. However, experience gained in the areas of fracture toughness of materials and pre-existing undetected defects shows that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, use of the allowances of ASME Code Case N–640 in developing the P/T limits will provide adequate protection against nonductile-type fractures of the RPV.

Development of the revised Monticello P/T limits was performed using the approved methodologies of 10 CFR 50, Appendix G, and using the allowances of ASME Code Case N–640. The P/T limit curves generated using these methods ensure the P/T limits will not be exceeded during any phase of reactor operation. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed change will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events.

Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

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

The proposed changes provide more operating margin in the P/T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with benefits being primarily realized during the pressure tests. Operation in the ''new'' regions of the newly developed P/T curves has been analyzed in accordance with the provisions of ASME Code, Section XI, Appendix G; 10 CFR 50 Appendix G, and ASME Code Case N–640, thus providing adequate protection against a nonductile-type fracture of the RPV.

The proposed changes do not alter any existing system relationships. The proposed changes do not result in any new or unanalyzed operation of any system or piece of equipment important to safety, and as a result, the possibility of a new type [of] event is not created.

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

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

As mentioned previously, the revised P/T limit curves provide more operating margin and thus, more operational flexibility than the current P/T limit curves. With the increased operational margin, a reduction in the safety margin results with respect to the existing curves. However, industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T limit curves are unrealistic and unnecessarily conservative. Accordingly, ASME Code Case N–640 takes into account the acquired knowledge and establishes more realistic methodologies for the development of P/T limit curves.

Use of ASME Code Case N–640 to develop the revised P/T curves utilized the K_{IC} fracture toughness curve in lieu of the K_{IA} curve as the lower bound for fracture toughness. Use of the K_{IC} curve to determine lower bound fracture toughness is more technically correct than using the K_{IA} curve. P/T curves based on the K_{IC} fracture toughness limits enhance overall plant safety by expanding the P/T window in the low-temperature operating region. The benefits which occur are a reduction in the duration of the pressure test and personnel safety while conducting inspections in primary containment with no decrease to the margin of safety. Therefore, operational flexibility is gained and an acceptable margin of safety to RPV non-ductile type fracture is maintained.

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 22, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to permit a one-time 5-year extension, to no later than March 2008, of the 10-year performance-based Type A test interval established in NEI 94–01, ''Nuclear Energy Institute Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' Revision 0, dated July 26, 1995.

This TS change has been prepared in accordance with the guidance provided in Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis.''

A plant-specific, risk-based evaluation has been performed in support of this one-time exception to extend the Type A test interval. This evaluation uses the latest Monticello probabilistic safety assessment (PSA) models to estimate the changes in risk associated with increasing the Type A testing interval. This risk assessment is consistent with current PSA best practices.

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

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

The proposed change to TS 4.7.A.2.b provides a one-time exception to the testing frequency for the Type A containment integrated leakage rate test. The current ten-year interval is based on past performance and the proposed change will only extend the Type A test frequency to fifteen years. The proposed change to the Technical Specifications does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the primary containment does not involve the prevention or identification of any precursors of an accident and therefore does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. The proposed change involves a one-time change to the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency specified in the Monticello Technical Specifications. As documented in NUREG–1493, ''Performance-Based Containment Leakage-Test Program,'' industry experience has shown that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment paths that are detected only by Type A tests is very small. An analysis of 144 integrated leak rate tests, including 23 failures, found that no failures were due to containment liner breach. NUREG–1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate tests to once per twenty years was found to lead to an imperceptible increase in risk. The Monticello risk-based evaluation of the proposed one-time extension to the Type A test frequency supports this conclusion. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the primary containment, combined with the containment inspections performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI and 10 CFR 50.65, Maintenance Rule, provide a high degree of assurance that the primary containment will not degrade in a manner that is detectable only by Type A tests and therefore does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change to Technical Specification 4.7.A.2.b involves a one-time exception to the current test interval for Type A containment leakage rate tests. The primary containment and the test requirements invoked to periodically demonstrate the integrity of the primary containment exist to ensure the ability to mitigate the consequences of an accident. Additionally, the reactor containment and its associated test requirements do not involve the prevention or identification of any precursors of an accident. The proposed change to the leakage rate test frequency does not involve any physical changes being made to the facility. In addition, the proposed extension of the Type A leakage rate test frequency does not change the operation of the plant such that a new failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created.

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

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

The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The proposed change involves only the extension of the interval between Type A containment leakage tests. The current interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. Type B and C containment leakage tests will continue to be performed at the frequency currently required by the plant Technical Specifications.

The NUREG–1493 generic study of the effects of extending containment leakage test intervals found that a twenty-year extension for Type A leakage tests resulted in an imperceptible increase in risk to the public. This study also found that, generically, the containment leakage paths are mainly detected by Type B and C tests. The proposed change involves a one-time extension of the frequency for Type A containment leakage tests; the overall primary containment leakage rate limit, specified by the Monticello Technical Specifications, is being maintained. The regular containment inspections being performed in accordance with the ASME Code, Section XI, and 10 CFR 50.65, Maintenance Rule, provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A tests. In addition, the containment monitoring capability that is inherent to boiling water reactors using an inert containment atmosphere allows for the detection of gross containment leakage that may develop during power operation. The cumulative effect of these inspections, tests and operating methods ensures that the margin of safety is maintained.

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 25, 2002.

Description of amendment request: The proposed amendment would revise the Monticello Technical Specifications (TSs) to allow the use of 10 CFR Part 50, Appendix J, Option B, for Types B and C containment leak rate testing. The proposed amendment would also revise the surveillance requirements (SRs) in TS 3.7/4.7 and provide a new TS Section 6.8.M, ''Primary Containment Leakage Rate Testing Program,'' in the ''Programs and Manuals'' section of the Monticello TSs. This proposed new TS program is formatted to be consistent with the NRC-approved guidance provided in Option B of the Primary Containment Leakage Rate Testing Program included in NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 2, dated April 2001.

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

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

The proposed changes deal exclusively with testing of features related to containment isolation. The changes only affect testing frequency and methodology. Containment leakage is not considered as an initiator of any accident previously evaluated.

Additionally, the proposed changes do not impact current plant operations or the design function of any system or component. The proposed changes do not change any accidents previously evaluated in the updated safety analysis report.

The proposed changes only affect the frequency of testing the containment penetrations and containment isolation valves. The proposed changes will allow test intervals to be extended in accordance with program requirements and 10 CFR Part 50, Appendix J, Option B, with reference to Regulatory Guide 1.163, and NEI 94–01, Rev. 0. The change in risk resulting from the proposed change, was evaluated by the NRC in the rule making process for implementing the Option B requirements, and are characterized in NUREG–1493. For Type B and C tests, the NRC concluded that the extension of test intervals as allowed by Option B would lead to only minor increases in potential offsite dose consequences.

The performance of the leakage tests themselves is not an input or consideration in any accident previously evaluated, thus the proposed change will not increase the probability of any such accident occurring. The same operability requirements remain in place for the primary containment, therefore, the consequences of an accident are not significantly increased. The proposed revision does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does it affect any assumptions or conditions in the accident analysis.

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

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

The proposed changes deal exclusively with testing of features related to containment isolation. The changes only affect testing frequency and methodology. The proposed changes to the TS will not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, and no changes to any equipment's function or the method of operating the equipment are being made. Since the proposed changes would not change the design, configuration or operation of the plant, they would not cause the containment leak rate testing to become an accident initiator. No new or different kinds of accident modes are created.

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

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

The proposed changes deal exclusively with testing of features of [sic] related to containment isolation. The changes only affect testing frequency and methodology. Containment leakage is not considered as an initiator of any accident previously evaluated.

The proposed changes do not exceed or alter a design basis or safety limit. The proposed changes only affect the methodology and frequency of Type B and C testing. The proposed performance based approach, provided by using Option B to 10 CFR Part 50, Appendix J, would continue to ensure that the containment leakage rates would not exceed the maximum allowable leakage rates defined in the Technical Specifications and assumed in the accident analysis.

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

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: June 11, 2002.

Description of amendment request: The proposed amendment would revise TS 3.1.8, ''Physics Test Exceptions,'' to correct a typographical error in the numbering of a function. The existing typographical error inappropriately makes the TS more restrictive than intended.

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.

The primary purpose of the Mode 2 Physics Tests exceptions is to permit relaxations of existing LCOs [limiting conditions for operation] to allow certain Physics Tests to be performed. The proposed change will permit the number of required channels specified in LCO 3.3.1, ''RPS [Reactor Protection System] Instrumentation,'' for Power Range Neutron Flux, P–10 interlock, to be reduced to ''3'' required channels for Physics Tests, as originally analyzed and approved by NRC. LCO 3.1.8 already allows one power range neutron flux channel to be bypassed, reducing the number of required channels from ''4'' to ''3''. With this reduction in the number of required channels, the fuel design criteria are preserved as long as the power level is limited to ≤5% RTP [rated thermal power], the reactor coolant temperature is kept ≥530°F, and shutdown margin (SDM) is within the limits provided in the Core Operating Limits Report (COLR). These three conditions are not affected by the proposed change. This change only restores the allowance previously analyzed as acceptable.

Therefore, the probability or consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

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 proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. This change does not introduce any new or different normal operation or accident initiators. With the reduction in the number of required instrumentation channels, the fuel design criteria continue to be preserved as originally analyzed.

Equipment important to safety will continue to operate as designed. The changes do not result in any event previously deemed incredible being made credible. The changes do 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.

The primary purpose of the Mode 2 Physics Tests exceptions is to permit relaxations of existing LCOs to allow certain Physics Tests to be performed. The analysis for Physics Tests is based on one power range neutron flux channel being bypassed. Therefore, reducing the requirement for an interlock associated with the bypassed channel is bounded by the original analysis. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components important to safety. Therefore, the proposed change will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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.

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

Date of amendment request: July 22, 2002.

Description of amendment request: The proposed amendment removes the reference to a specific computer program for monitoring core radial peaking factors when a core power tilt is present. Instead, the functional requirement is specified. These changes clarify the requirements for core tilt monitoring associated with a computer system upgrade and changes in computer programs. Also, it is proposed to add clarification in the Basis section for Technical Specification (TS) 2.10.4 regarding the application of TS 2.10.4(1)(b) when the plant computer incore detector alarms for monitoring core linear heat rate become inoperable.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change does not result in any changes to the existing core power distribution monitoring requirements. There is no change in the analysis values used in the evaluation of the transients and accidents. All of the evaluated transients and accidents currently show acceptable results and will not be affected by this change. Incorporating this change will not affect the probability of an accident, since core power distribution monitoring is not changed. The change to the wording of the core power distribution monitoring specifications will not change the failure possibilities for reactor protective features. The effect of the proposed change is the clarification of the existing core power distribution monitoring requirements.

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

The change to the wording of the core power distribution monitoring specifications does not provide the possibility of the creation of a new or different type of accident. Changing the wording of the core power distribution monitoring specifications does not change the method of core power distribution monitoring or the expected response of reactor protective features. The reactor will operate within previously analyzed limits.

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

The proposed change to the wording of the core power distribution monitoring specifications does not constitute a significant reduction in the margin of safety due to the core power distribution monitoring requirements are not changed and are consistent with the assumptions contained in the transient and accident analyses contained in the Updated Safety Analysis Report shown to produce acceptable results.

The acceptance criteria used in the analysis have been developed for the purpose of use in design basis accident analyses such that meeting these limits demonstrates adequate protection of public health and safety. An acceptable margin of safety is inherent in these licensing limits. Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: July 22, 2002.

Description of amendment request: The proposed amendment deletes technical specification (TS) requirements for missed surveillances from TS 3.0.4 and adds TS 3.0.5 for missed surveillances consistent with the Improved Standard Technical Specification (ITS) for Combustion Engineering Plants, NUREG–1432, Revision 2, and Technical Specification Task Force Change Traveler TSTF–358, Revision 6. This proposed amendment also adds a TS requirement for a Bases Control Program consistent with that presented in Section 5.5 of the ITS (NUREG–1432, Revision 2), in accordance with the guidance published in the **Federal Register** on September 28, 2001, ''Notice of Availability of Model Application Concerning Technical Specification Improvement to Modify Requirements Regarding Missed Surveillances Using the Consolidated Line Item Improvement Process,'' (66 FR 49714). Appropriate TS Bases changes are also provided in accordance with the Consolidated Line Item Improvement Process.

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

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

The proposed change to incorporate Improved Standard Technical Specification (ITS) SR 3.0.3 relaxes the time allowed to perform a missed Surveillance. The time between Surveillances is not an initiator to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to provide a Technical Specification (TS) Bases Control Program presents more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to incorporate ITS SR 3.0.3 does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to provide a TS Bases Control Program presents more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed change to provide a TS Bases Control Program presents more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: July 23, 2002.

Description of amendment request: This proposed amendment will: (1) Remove the requirement to demonstrate operability of redundant auxiliary feedwater system components, and (2) provide an allowed outage time to restore operability of the emergency feedwater storage tank. Each of the revisions is modeled after the Improved Standard Technical Specifications.

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

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

The proposed changes to Technical Specifications Sections 2.5 establish an allowed outage time and actions required for restoring operability. The proposed Technical Specifications address the regulatory requirements for equipment required for Auxiliary Feedwater Systems per NUREG–0635 [''NRC Requirements for Auxiliary Feedwater Systems'']. The change will ensure that proper Limiting Conditions for Operation are entered for equipment or functional inoperability. There are no physical alterations being made to the Auxiliary Feedwater System or related systems. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not result in any physical alterations to the Auxiliary Feedwater System, any plant configuration, systems, equipment, or operational characteristics. There will be no changes in operating modes, or safety limits, or instrument limits. With the proposed changes in place, Technical Specifications will retain requirements for the Auxiliary Feedwater System. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes clarify the regulatory requirements for the Auxiliary Feedwater System as defined by NUREG–0635 and NUREG–0737. The times established are identical to those invoked by the present Technical Specifications or to those previously reviewed and approved for use by the NRC. The proposed changes will not alter any physical or operational characteristics of the Auxiliary Feedwater System and associated systems and equipment. Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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: July 25, 2002.

Description of amendment request: The proposed amendments would change the Susquehanna Steam Electric Station Final Safety Analysis Report by revising the Reactor Pressure Vessel (RPV) Material Surveillance Program. Specifically, the licensee proposes to replace the current plant-specific RPV material surveillance program with the Boiling Water Reactor (BWR) Integrated Surveillance Program.

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

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

No.

The proposed change implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50 [Title 10 of the CODE OF FEDERAL REGULATIONS, Part 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 integrity. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

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

Attorney for licensee: 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: June 28, 2002.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) by relaxing the secondary containment requirements and eliminating the Filtration, Ventilation, and Recirculation system (FRVS) charcoal filters.

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 definition of CORE ALTERATIONS has been revised to define that control rod movement, provided there are no fuel assemblies in the associated core cell is not a core alteration. This is consistent with Standard Technical Specifications (STS) NUREG–1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The TS presently provide[s] a period of 7 days to restore an inoperable FRVS ventilation unit when performing activities with the potential for draining the reactor vessel or discontinue such activities. Operation of the redundant train will ensure that the remaining subsystem is operable, that no failures, which could prevent automatic actuation, have occurred and that any other failures will be readily detected. This is consistent with STS, NUREG–1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The proposed changes associated with the FHA [fuel handling accident] do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report (UFSAR). The FHA for the HCGS is defined as a drop of a fuel assembly over irradiated assemblies in the reactor core 24 hours after reactor shutdown. AST [accident source term] is used to evaluate the dose consequences of a postulated accident. The FHA has been analyzed without credit for Secondary Containment, Filtration Recirculation and Ventilation System (FRVS), and Control Room Emergency Filtration (CREF) system. The resultant radiological consequences are within the acceptance criteria set forth in 10 CFR 50.67 and Regulatory Guide [(RG)] 1.183. This amendment does not alter the methodology or equipment used directly in fuel handling operations. The equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator. Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

The Loss of Coolant Accident (LOCA) Dose Calculation has been revised to (1) eliminate credit for the FRVS recirculation charcoal filters, (2) reduce credited efficiency of FRVS vent charcoal filters, (3) reduce Engineered Safety Feature (ESF) leakage from 10 gpm to 1 gpm and (4) reduce control room unfiltered in-leakage to 350 cfm.

These proposed changes do not eliminate any safety system. The changes are only associated with the credit provided by the system in reducing the radiological consequences and therefore, do not affect any accident initiator. The results of that analysis show that the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses are of the same order of magnitude as the previous analysis and remain within the acceptance criteria in 10 CFR 50.67 and Regulatory Guide 1.183.

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 amendment will not create the possibility for a new or different type of accident from any accident previously evaluated. Changes to the allowable activity in the primary and secondary systems do not result in changes to the design or operation of these systems. The evaluation of the effects of the proposed changes indicates that all design standard and applicable safety criteria limits are met.

Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems.

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 changes revise the TS to establish operational conditions where specific activities represent situations during which significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the regulatory guidelines. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed TS continue[s] to ensure that the TEDE [total effective dose equivalent] for the CR, the EAB, and LPZ boundaries are below the corresponding acceptance criteria specified in 10 CFR 50.67 and RG1.183.

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, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: Jacob Zimmerman, Acting.

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

Date of amendment request: July 25, 2002.

Brief description of amendments: The proposed amendments would change the CPSES Facility Operating Licenses as follows: Section 2.C.(4)(b) would be changed to be consistent with the license conditions stated in the U.S. Nuclear Regulatory Commission (NRC) Order and Safety Evaluation issued December 21, 2001, which approved the direct transfer of ownership interest and operating authority for CPSES to TXU Generation Company LP; Section 2.E which requires reporting any violations of the requirements contained in Section 2.C of the licenses would be deleted. Additionally, Technical Specification Table 5.5–2 ''Steam Generator Tube Inspection,'' Table 5.5–3, ''Steam Generator Repaired Tube Inspection for Unit 1 Only,'' and Section 5.6.10, ''Steam Generator Tube Inspection Report,'' would be revised to delete the requirement to notify the NRC pursuant to 50.72(b)(2) of Title 10 of the Code of Federal Regulations (10 CFR) if the steam generator tube inspection results are in a C–3 classification.

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

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

Response: No.

The requested change to revise Section 2.C.(4)(b) of the Operating Licenses is consistent with NRC Order and Safety Evaluation approved December 21, 2001 for Facility Operating Licenses NPF–87 and NPR–89. The requested change to delete Section 2.E of the Operating Licenses and the changes to revise Technical Specification Table 5.5–2, Table 5.5–3 and Section 5.6.10 are consistent with the changes recently implemented in 10 CFR 50.72 and 10 CFR 50.73.

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (*i.e.*, fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves administrative changes only.

No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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.

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

Date of amendment request: July 26, 2002.

Brief description of amendment request: The proposed amendments would amend Operating Licenses DPR–58 and DPR–74 to add a license condition allowing a one-time 140-hour allowed outage time for the essential service water (ESW) system, to allow ESW pump replacement during plant operation.

Date of publication of individual notice in ***Federal Register:*** August 8, 2002 (67 FR 51603).

Expiration date of individual notice: September 9, 2002.

Florida Power and Light Company, Docket No. 50–251, Turkey Point Plant, Unit 4, Miami-Dade County, Florida

Date of amendment request: July 29, 2002.

Description of amendment request: Revised Technical Specifications to allow use of an alternate method of determining rod position for a control rod with an inoperable rod position indication. Effective during the current operating cycle until repair of the indication system can be completed at the next outage.

Date of publication of individual notice in the ***Federal Register:*** August 2, 2002 (67 FR 50473).

Expiration date of individual notice: August 16, 2002.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Dominion Nuclear Connecticut, Inc. et al., Docket Nos. 50–245, 50–336, and 50–423 Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London County, Connecticut

Date of amendment request: August 8, 2001.

Brief description of amendment: The amendment incorporates two changes into each operating license. The physical protection (security) related license condition is revised to indicate that the physical security program plans listed, may, rather than do, contain safeguards information; and the plant name is changed from the ''Millstone Nuclear Power Station'' to the ''Millstone Power Station.''

Date of issuance: August 8, 2002.

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

Amendment Nos.: Unit 1,–110, Unit 2–269, and Unit 3–208.

Facility Operating License Nos. DPR–21, DPR–65 and NPF–49: The amendment revised the operating licenses.

Date of initial notice in ***Federal Register:*** October 17, 2001 (66 FR 52798). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: October 1, 2001, as supplemented by letters dated June 26, and August 5, 2002.

Brief description of amendment: The amendment will revise the Technical Specifications (TSs) limiting condition for operation and surveillance requirements associated with verification of reactor coolant system operational leakage. Conforming changes are also made to the associated TS Bases.

Date of issuance: August 21, 2002.

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

Amendment No.: 209.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 14, 2001 (66 FR 57120). The supplements dated June 26 and August 5, 2002, were within the scope of the original application as published in the **Federal Register** and did not change the staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 19, 2002.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.10, ''Technical Specification (TS) Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999, that became effective March 13, 2001.

Date of issuance: August 15, 2002.

Effective date: August 15, 2002.

Amendment No.: 177.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42821). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 15, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 2, 2001, as supplemented January 9 and July 10, 2002.

Brief description of amendment: The amendment changes the Current Technical Specifications and the Improved Technical Specifications Main Steam Isolation Valve Leakage Surveillance Requirement. The licensee will also make conforming changes to the associated Bases and the Primary Containment Leakage Rate Testing Program.

Date of issuance: August 13, 2002.

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

Amendment No.: 275.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2923). The January 9 and July 10, 2002, letters provided clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination as published in the **Federal Register**. The January 9 supplement also corrected the original application date from November 2, 2000, to November 2, 2001.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 2002.

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: August 22, 2001, as supplemented on March 5, 2002.

Brief description of amendment: The amendment revises the Technical Specification (TS) Surveillance Requirement (SR) 3/4.7.B.1.a.2 for the Standby Gas Treatment (SBGT) System and the associated TS Bases 3/4.7.B.1, by increasing the SBGT inlet heaters minimum output testing requirement from 14 kW to 20 kW.

Date of issuance: August 20, 2002.

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

Amendment No.: 194.

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

Date of initial notice in ***Federal Register:*** November 14, 2001 (66 FR 57121). The supplement dated March 5, 2002, provided additional information that clarified the application, and did not expand the scope of the application or change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 20, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 20, 2001, as supplemented on February 13, 2002.

Brief description of amendment: The amendment to the Technical Specifications (TSs) revises certain requirements associated with demonstrating the operability of alternate trains when redundant equipment is made or found to be inoperable. The TSs revised include: 4.4.B, 4.5.A.2, 4.5.A.3, 4.5.A.4, 4.5.B.2, 4.5.C.2, 4.5.C.3, 4.5.D.2, 4.5.D.3, 4.5.E.2, 4.5.F.2, 4.5.H.1, 4.7.B.3.c, 4.10.B.1, 4.10.B.3.b.2. Some format and typographical errors were also corrected.

Date of Issuance: August 14, 2002.

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

Amendment No.: 209.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 19, 2001 (66 FR 48292). The February 13, 2002, supplement was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 14, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 20, 2001, as supplemented on March 28, 2002.

Brief description of amendment: This amendment moves Table 4.7.2, "Primary Containment Isolation Valves" and references, to the Technical Requirements Manual; changes surveillance requirement 4.7.B.1.b to reflect that the Standby Gas Treatment system duct heater needs to meet relative humidity design-basis requirements; adds Section 3.7.E, "Reactor Building Automatic Ventilation System Isolation Valves," to the Table of Contents; removes wording in 3.5.A.4.a and b referencing a one-time 30-day Limiting Condition for Operation; and, makes administrative changes to Sections 5.3 and 6.4.

Date of Issuance: August 21, 2002.

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

Amendment No.: 210.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 26, 2001 (66 FR 66474). The March 28, 2002, supplemented was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 251, Turkey Point Plant, Unit 4, Miami-Dade County, Florida

Date of amendment request: July 29, 2002, as supplemented August 14 and August 16, 2002.

Description of amendment request: The amendment revised Technical Specifications 3/4.1.3.1, 3/4.1.3.2 and 3/4.1.3.5 to allow the use of an alternate method of determining rod position for the control rod C–9, until the end of Cycle 20 or until repairs can be conducted on the Analog Rod Indication System at the next outage of sufficient duration, whichever comes first.

Date of issuance: August 20, 2002.

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

Amendment No.: 216.

Facility Operating License No. (DPR–41): Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. August 2, 2002 (67 FR 50473). The licensee's August 14 and August 16, 2002, submittals of supplemental information did not affect the original no significant hazards consideration determination, and did not expand the scope of the request as noticed on August 2, 2002. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by August 16, 2002, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated August 20, 2002.

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: June 7, 2002.

Brief description of amendment: The amendment deletes Section 5.5.3, ''Post Accident Sampling,'' from the Technical Specifications and thereby eliminates the requirements to have and maintain the Post Accident Sampling System.

Date of issuance: August 9, 2002.

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

Amendment No.: 106.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 9, 2002 (67 FR 45570). The staff's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 2, 2002.

Brief description of amendments: These amendments revised 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: August 12, 2002.

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

Amendment Nos.: 205 and 179.

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

Date of initial notice in ***Federal Register:*** May 28, 2002 (67 FR 36932). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 12, 2002.

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: March 29, 2002.

Brief description of amendment: The amendment allows the use of the current pressure-temperature (P–T) limit curves through Cycle 12. The amendment also removes notes from the Technical Specifications that state that the curves are valid for 32 effective full power years.

Date of issuance: August 13, 2002.

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

Amendment No.: 139.

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

Date of initial notice in ***Federal Register:*** May 14, 2002 (67 FR 34491). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 2002.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: October 21, 2001, as supplemented by letters dated February 11 and May 27, 2002.

Brief description of amendments: The amendments revised the Technical Specifications, Table 3.3.1–1, ''Reactor Trip System Instrumentation'' and associated Bases B 3.3.1. A limit or ''clamp'' on the Overtemperature Delta Temperature reactor trip function addresses design issues related to fuel rod design under transient conditions. In addition, editorial revisions to bases B 3.3.1 are included.

Date of issuance: August 9, 2002.

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

Amendment Nos.: 127 & 105.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15608). The supplements dated February 11, 2002, and May 27, 2002, provided clarifying information that did not change the scope of the October 30, 2001, 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 August 9, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 26, 2001, as supplemented by letters dated February 4 and June 12, 2002.

Brief description of amendments: The amendments revise TS 5.5.16, ''Containment Leakage Rate Testing Program'' to extend the 10 CFR Part 50, Appendix J, Type A, Containment Integrated Leak Rate Test date for Comanche Peak Steam Electric Station, Units 1 and 2, from the fall of 2002 to December 2008 for Unit 1, and from the fall of 2006 to December 2012 for Unit 2. The following phrase implements this change in TS 5.5.16.a: '' * * * as modified by the following exception: 1. NEI 94–01—1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993, Type A Test (Unit 1) and the December 1, 1997, Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2).''

Date of issuance: August 15, 2002.

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

Amendment Nos.: 98 and 98.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 5, 2002 (67 FR 5340). The February 4 and June 12, 2002, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 23rd day of August 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–22197 Filed 8–30–02; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

September 12, 2002 Board of Directors Meeting

Time and Date: Thursday, September 12, 2002, 1:30 p.m. (Open Portion), 1:45 p.m. (Closed Portion).

Place: Offices of the Corporation, Twelfth Floor Board Room, 1100 New York Avenue, NW., Washington, DC.

Status: Meeting open to the Public from 1:30 p.m. to 1:45 p.m., Closed portion will commence at 1:45 p.m. (approx.).

Matters to be Considered:

1. President's Report
2. Approval of May 22, 2002 Minutes (Open Portion)

Further Matters to be Considered: (Closed to the Public 1:45 p.m.)

1. Proposed FY 2004 Budget Proposal and Allocation of Retained Earnings
2. Finance Project in Russia, Azerbaijan, Uzbekistan, Kazakhstan, and Ukraine
3. Finance Project—Global
4. Approval of May 22, 2002 Minutes (Closed Portion)
5. Pending Major Projects
6. Reports

Contact Person for Information: Information on the meeting may be obtained from Connie M. Downs at (202) 336–8438.

Dated: August 29, 2002.

Connie M. Downs,

Corporate Secretary, Overseas Private Investment Corporation.

[FR Doc. 02–22524 Filed 8–29–02; 2:13 pm]

BILLING CODE 3210–01–M

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

Notice is hereby given, pursuant to the provisions of the Government in the Sunshine Act, Public Law 94–409, that the Securities and Exchange Commission will hold the following meeting during the week of September 2, 2002:

A Closed Meeting will be held on Tuesday, September 3, 2002, at 10 a.m.

Commissioner Campos, as duty officer, determined that no earlier notice thereof was possible.

Commissioners, Counsel to the Commissioners, the Secretary to the Commission, and recording secretaries will attend the Closed Meeting. Certain staff members who have an interest in the matters may also be present.

The General Counsel of the Commission, or his designee, has certified that, in his opinion, one or more of the exemptions set forth in 5 U.S.C. 552b(c)(3), (5), (7), (9)(B) and (10) and 17 CFR 200.402(a)(3), (5), (7), (9)(ii) and (10), permit consideration of the scheduled matters at the Closed Meeting.

The subject matter of the Closed Meeting scheduled for Tuesday, September 3, 2002, will be:

Formal orders of investigation;
Institution and settlement of injunctive actions; and
Institution and settlement of administrative proceedings of an enforcement nature.

At times, changes in Commission priorities require alterations in the scheduling of meeting items. For further information and to ascertain what, if any, matters have been added, deleted or postponed, please contact: The Office of the Secretary at (202) 942–7070.

Dated: August 29, 2002.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 02–22503 Filed 8–29–02; 11:37 am]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–46419; File No. SR–NASD–2002–109]

Self-Regulatory Organizations; Notice of Filing of a Proposed Rule Change by the National Association of Securities Dealers, Inc. Relating to Fees for Nasdaq's InterMarket

August 27, 2002.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (''Act''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on August 8, 2002, the National Association of Securities Dealers, Inc. (''NASD'' or ''Association''), through its subsidiary, the Nasdaq Stock Market, Inc. (''Nasdaq''), filed with the Securities and Exchange Commission (''Commission'' or ''SEC'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by Nasdaq. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons.

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

[2] 17 CFR 240.19b–4.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

Nasdaq proposes to: (i) Modify the execution fees for Nasdaq InterMarket trades executed through the Intermarket Trading System (''ITS'') and Nasdaq's Computer Assisted Execution System (''CAES''); and (ii) establish a credit for the liquidity provider for executions via ITS and CAES.[3] Nasdaq will implement the proposed rule change as quickly as practicable following approval. Below is the text of the proposed rule change. Proposed new language is in italics; proposed deletions are in brackets.

* * * * *

7010. System Services

(a)–(c) No change.

(d) Computer Assisted Execution Service.

The charges to be paid by members receiving the Computer Assisted Execution Service (CAES) shall consist of a fixed service charge and a per *share* transaction charge plus equipment-related charges.

(1) Service Charges

$100 per month for each market maker terminal receiving CAES.

(2) Transaction Charges

(A) [As of January 1, 1998, $0.50 per execution] *$0.003 per share executed up to a maximum of $75 per execution* shall be paid by an order entry firm or CAES market maker that enters an order into CAES that is executed in whole or in part*, and $0.002 per share executed up to a maximum of $50 per execution shall be credited to the CAES market maker that executes such an order.*[*]

(B) [As of November 1, 1997, $1.00 per commitment] *$0.002 per share executed up to a maximum of $75 per execution* shall be paid by any member that sends a commitment through the ITS/CAES linkage to buy or sell a listed security that is executed in whole or in part*, and $0.001 per share executed up to a maximum of $35 per execution shall be credited to a member that executes such an order.*[**]

[3] On June 13, 2002, the NASD, through its subsidiary, Nasdaq, filed a similar proposed rule change that was effective upon filing pursuant to Section 19(b)(3)(A) of the Act. 15 U.S.C. 78s(b)(3)(A). *See* Securities Exchange Act Release No. 46153 (July 1, 2002), 67 FR 45164 (July 8, 2002) (SR–NASD–2002–68). The proposal was summarily abrogated by Commission order on July 2, 2002. *See* Securities Exchange Act Release No. 46159.