facility meets the license termination criteria in subpart E of 10 CFR part 20.

III. Finding of No Significant Impact

The NRC staff has evaluated Praelux's request and the results of the surveys and has concluded that the completed action complies with 10 CFR part 20. The staff has prepared the EA (summarized above) in support of the proposed license amendment to terminate the license and release the facility for unrestricted use. On the basis of the EA, the NRC has concluded that the environmental impacts from the proposed action are expected to be insignificant and has determined not to prepare an environmental impact statement for the proposed action.

IV. Further Information

The EA and the documents related to this proposed action, including the application for the license amendment and supporting documentation, are available for inspection at NRC's Public Electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html* (ADAMS Accession Nos. ML031680934, and ML031350739. These documents are also available for inspection and copying for a fee at the Region I Office, 475 Allendale Road, King of Prussia, PA 19406. Any questions with respect to this action should be referred to Kathy Modes, Nuclear Materials Safety Branch 2, Division of Nuclear Materials Safety, Region I, 475 Allendale Road, King of Prussia, Pennsylvania, 19406, telephone (610) 337–5251, fax (610) 337–5269.

Dated in King of Prussia, Pennsylvania this 17th day of June, 2003.

For the Nuclear Regulatory Commission.

John D. Kinneman,

Chief, Nuclear Materials Safety Branch 2, Division of Nuclear Materials Safety, Region I.

[FR Doc. 03–15857 Filed 6–23–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Notice

AGENCY: Nuclear Regulatory Commission.

DATES: Weeks of June 23, 30, July 7, 14, 21, 28, 2003.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED: emsp;

Week of June 23, 2003

There are no meetings scheduled for the Week of June 23, 2003.

Week of June 30, 2003–Tentative

Tuesday, July 1, 2003

10 a.m.—Briefing on Status of Office of Nuclear Security and Incident Response (NSIR) Programs, Performance, and Plans (Closed—Ex. 1).

Week of July 7, 2003—Tentative

There are no meetings scheduled for the Week of July 7, 2003.

Week of July 14, 2003—Tentative

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

Week of July 21, 2003—Tentative

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

Week of July 28, 2003—Tentative

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

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

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

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

Dated: June 19, 2003.

D.L. Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–16004 Filed 6–20–03; 11:21 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, May 30, 2003, through June 12, 2003. The last biweekly notice was published on June 10, 2003 (68 FR 28844).

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

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

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

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

Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By July 24, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's

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

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: May 12, 2003.

Description of amendments request: The proposed amendment would extend several Required Action Completion times for inoperable diesel generators (DGs) identified in Technical Specification 3.8.1, ''AC Sources-Operating.''

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 proposed Technical Specification changes do not affect the design, operational characteristics, function or reliability of the DGs. The DGs are not accident initiators, and extending the DG Required Action Completion Times will not impact the frequency of any previously evaluated accidents. The design basis accidents will remain the same postulated events described in the Updated Final Safety Analysis Report. In addition, extending the DG Required Action Completion Times will not impact the consequences of an accident previously evaluated. The consequences of previously evaluated accidents will remain the same during the proposed extended Required Action Completion Times as during the current Required Action Completion Times. The ability of the remaining DGs to mitigate the consequences of an accident will not be affected since no additional failures are postulated while equipment is inoperable within the Technical Specification Required Action Completion Times. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The duration of a Technical Specification Required Action Completion Time is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. A risk informed assessment was performed that concluded that the plant risk is acceptable and consistent with the guidance contained in Regulatory Guide 1.177.

The additional proposed changes to renumber action requirements and the correction of a misspelled word will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications do not impact any system or component in a manner that could cause an accident. The proposed changes will not alter the plant configuration or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operator following an accident will not be significantly different. In addition, the proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided by the DGs is to provide emergency back-up power supply to systems required to mitigate the consequences of postulated accidents. The engineered safety features systems on either of the two trains for each unit provide for the minimum safety functions necessary to shutdown the units and maintain it in a safe shutdown condition. Each of the two trains can be powered from one of the offsite power sources or its associated DG. In addition, the 0C DG (Station Blackout DG) is available to provide power to any of the trains. This design provides adequate defense in-depth to ensure that diverse power sources are available to accomplish the required safety functions. Thus, with a safety-related DG out-of-service, there is sufficient means to accomplish the safety functions and prevent the release of radioactive material in the event of an accident.

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

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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.

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

Date of amendment request: February 26, 2003.

Description of amendment request: The proposed amendments would allow the licensee to revise the Updated Final Safety Analysis Report to include a description of a load drop analysis performed for handling reactor cavity shield blocks weighing greater than 110 tons with the Dresden, Units 2 and 3, reactor building crane.

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

The proposed change will allow use of a load drop analysis performed for handling the reactor cavity shield blocks weighing greater than 110 tons with the reactor building crane during power operation. The load drop analysis demonstrates that dropping a reactor cavity shield block within the designated safe load path from the heights assumed in the analysis will not affect the capability of safety-related equipment to perform its function.

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

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

The proposed change will allow use of a load drop analysis performed for handling the reactor cavity shield blocks weighing greater than 110 tons with the reactor building crane during power operation. The load drop analysis demonstrates that dropping a reactor cavity shield block within the designated safe load path from the heights assumed in the analysis will not affect the capability of safety-related equipment to perform its function. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will allow use of a load drop analysis performed for handling the reactor cavity shield blocks weighing greater than 110 tons with the reactor building crane during power operation. The load drop analysis demonstrates that dropping a reactor cavity shield block within the designated safe load path from the heights assumed in the analysis will not affect the capability of safety-related equipment to perform its function. Therefore, it is concluded that the proposed change does not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 19, 2003.

Description of amendment request: The proposed amendments would revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change will decrease the frequency associated with TS Surveillance Requirement (SR) 3.7.7.1 for Turbine Bypass Valve (BPV) testing from 7 to 31 days. The proposed change is consistent with the testing frequency contained in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001, for BPV testing.

The 7-day frequency associated with SR 3.7.7.1 was established in the LaSalle County Station (LSCS) TS during conversion to Improved Technical Specifications (ITS) format due to the testing frequency contained in the LSCS custom TS and the difficulties experienced with other Electro-Hydraulic Control (EHC) system valves to consistently pass their surveillance tests. LSCS has recently re-evaluated the performance of these valves and has determined that the current performance of these valves supports decreasing the testing frequency of the BPVs from 7 to 31 days.

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

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

The proposed change will decrease the frequency associated with Surveillance Requirement (SR) 3.7.7.1 for turbine bypass valve (BPV) testing from 7 to 31 days. The proposed change is consistent with the testing frequency contained in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001, for BPV testing. The performance of BPV surveillance testing is not a precursor to any accident previously evaluated.

Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The Main Turbine Bypass System is required to be operable to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit Minimum Critical Power Ratio (MCPR) is not exceeded. An operable Main Turbine Bypass System requires the BPVs to open in response to increasing main steam line pressure. The performance of BPVs surveillance testing provides assurance that the valves will operate as assumed in accidents previously evaluated. Thus, the radiological consequences of any accident previously evaluated are not increased.

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 change does not affect the control parameters governing unit operation and does not introduce any new equipment, modes of system operation or failure mechanisms. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change will decrease the frequency associated with SR 3.7.7.1 for BPV testing from 7 to 31 days. The proposed change is consistent with the BPV testing frequency contained in NUREG–1433, Revision 2, and does not affect the design parameters or the setpoints associated with BPV operation. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania

Date of amendment request: March 26, 2003.

Description of amendment request: The proposed amendments would modify Technical Specifications (TSs) 4.0.1 and 4.0.3 to be consistent with the Improved Standard Technical Specifications. The proposed amendments would also modify the TS requirements for missed surveillances in TS 4.0.3 to be consistent with the Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF), Standard Technical Specification Change TSTF–358, Revision 6.

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 that portion of the following NSHC determination, related to the adoption of the TSTF–358, Revision 6, changes to the TSs in its application dated March 26, 2003.

Basis for proposed no significant hazards consideration determination:

Item 1: Modification of TSs 4.0.1 and 4.0.3 to be consistent with the Improved Standard Technical Specifications.

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

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

Response: No.

The proposed changes involve rewording of the existing Technical Specifications to be consistent with NUREG–1431, Revision 2. These modifications involve no technical changes to the existing Technical Specifications. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

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

Response: No.

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

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

Response: No.

The proposed changes involve rewording of the existing Technical Specifications to be consistent with NUREG–1431, Revision 2. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, there will be no reduction in a margin of safety.

Item 2: Incorporation of TSTF–358—Revision 6.

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards on 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.

The NRC staff has reviewed the licensee's analysis of Item 1 and the licensee's reference to the analysis included in the consolidated line-item improvement process **Federal Register** Notice, June 14, 2001 (66 FR 32400) for Item 2, and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: November 30, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by decreasing the pressurizer high level limit and by revising the required action when the pressurizer is inoperable.

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

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

Response: No.

The new pressurizer high level limit is more restrictive than the existing limit, and accident initial conditions, probability, and assumptions remain as previously analyzed. The proposed change to the pressurizer allowed outage time will have no significant effect on accident initiation frequency. The proposed changes do not invalidate the assumptions used in evaluating the radiological consequences of any accident. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

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

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

Response: No.

The proposed change to the pressurizer high level limit will ensure an adequate margin of safety is maintained. The proposed change to the pressurizer allowed outage time is minimal and will not have a significant effect on any margin of safety. 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 8, 2002.

Description of amendment request: Omaha Public Power District (OPPD) has proposed the following changes to the Technical Specifications (TSs): (1) Use a pressure temperature limits report (PTLR), (2) change the minimum boltup temperature, (3) modify the TSs to reflect the revised low temperature overpressure protection (LTOP) methodology and analysis that is submitted for review and approval, (4) perform LTOP analyses ''in-house,'' (5) change the LTOP enable temperature, (6) modify TS 2.10.1 to exactly specify the reactor coolant system (RCS) temperature at which the reactor can be made critical, and (7) add a TS for a maximum pressure value for the safety injection tanks. The use of a PTLR requires the relocation of TS Figure 2–1 (RCS Pressure—Temperature Limits for Heatup, Cooldown, and In-service Test) into Figure 5–1 of the PTLR. As a result of these changes, the following TSs are required to either be modified or added: define the PTLR in Definitions; TS 2.1.1(8); TS 2.1.1(11); Basis Section of TS 2.1.1; TS 2.1.2, including the TS 2.1.2 Basis and Reference Sections; TS 2.1.6(4); TS 2.3(1)(c); TS 2.3(3); TS 2.3 References; TS 2.10.1 and TS 2.10.1 Basis Section; Table 3–5, item 23, TS 3.3(1)(c); and TS 5.9.6.

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

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

The proposed changes will not increase the probability or consequence of any accident for the following reasons:

(1) The proposed changes relocate the Pressure—Temperature (P–T) limit curves and low temperature over pressure protection (LTOP) system setpoints to the Pressure and Temperature Limits Report (PTLR). Compliance with these curves and limits continues to be required by the Technical Specifications (TSs). Changes to the curves will be controlled by TS 5.9.6, which contains the NRC approved methodologies used in the development of the PTLR. The change to the P–T limit curve as shown on Figure 5–1 of the PTLR is in compliance with Reference 10.11 [of the licensee's October 8, 2002, submittal], Westinghouse Electric Company/Combustion Engineering's (W/CE's) methodology and ASME Code Case N–640 for performing P–T limit curves.

(2) Revisions to the LTOP system limits can only be made in accordance with the approved methodologies stated in TS 5.9.6 with any resulting setpoint changes controlled by the 10 CFR 50.59 process. The PTLR in combination with the limitations imposed by the TSs will ensure the integrity of the reactor vessel pressure boundary.

(3) The conservative, but lower minimum boltup temperature and LTOP enable temperature are in compliance with Reference 10.12 [of the licensee's October 8, 2002, submittal]. Since the P–T limit curves and LTOP analysis are analyzed to the same temperatures as these proposed temperature values, there is no reduction to the margin of safety.

(4) Restricting the RCS temperature at which the reactor can be made critical is more conservative than the minimum temperature requirements for core critical operations based on fracture mechanics considerations as required by Reference 10.11 [of the licensee's October 8, 2002, submittal] during physics testing.

(5) Addition of a maximum pressure to the safety injection tanks (SITs) ensures compliance with Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Therefore, the probability or consequence of any accident is not increased.

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

The proposed revision does not change any equipment required to mitigate the consequences of an accident. The continued use of the same TS administrative controls prevents the possibility of a new or different kind of accident. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident events are postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Relocating the P–T limit curves and LTOP system setpoints to the PTLR is in compliance with Reference 10.7 [of the licensee's October 8, 2002, submittal]. Future updates of the PTLR will be conducted under the 10 CFR 50.59 process utilizing NRC approved methodologies. Updating the P–T limit curve is in accordance with Reference 10.11 [of the licensee's October 8, 2002, submittal], W/CE's methodology and ASME Code Case N–640. Reduction of the minimum boltup temperature and LTOP enable temperature is in compliance with Reference 10.12 [of the licensee's October 8, 2002, submittal]. Restricting the reactor coolant system (RCS) temperature at which the reactor can be made critical is more conservative than the minimum temperature requirements for core critical operations based on fracture mechanics considerations as required by Reference 10.11 [of the licensee's October 8, 2002, submittal], during physics testing. Addition of a maximum pressure to the SITs is in accordance with Criterion 2 of 10 CFR 50.36(c)(2)(ii). Additionally, the LTOP methodology and analysis conforms to Reference 10.10 [of the licensee's October 8, 2002, submittal]. Therefore, the proposed changes do not involve a significant reduction to the margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: April 2, 2003.

Description of amendment requests: The proposed license amendments would revise Technical Specification (TS) 5.5.11, ''Ventilation Filter Testing Program (VFTP),'' to change the surveillance frequency, penetration, and relative humidity requirements for laboratory testing of the charcoal adsorber for the control room, auxiliary building, and fuel handling building ventilation systems. This would also eliminate the charcoal preheater testing requirements. TS 3.7.10, ''Control Room Ventilation System (CRVS),'' and TS 3.7.12, ''Auxiliary Building Ventilation System (ABVS),'' will also be revised to be consistent with these changes. These changes are in accordance with Regulatory Guide 1.52, Revision 3, ''Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,'' Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' and the requirements in American Society for Testing and Materials D3803–1989, ''Standard Technical Method for Nuclear-Grade Activated Carbon.'' In addition, TS 3.7.10 would be revised by adding a note allowing the control room boundary to be open intermittently under administrative control; adding a new required TS Action for two CRVS trains being inoperable due to an inoperable control room boundary, and revising the relettered Condition F to add ''for reasons other than Condition B.'' TS Surveillance Requirement (SR) 3.7.12.3 would be revised to limit its applicability and TS 3.7.13, ''Fuel Handling Building Ventilation System (FHBVS),'' would be revised to add the word ''recently'' to qualify the irradiated fuel in the statement of applicability. These proposed revisions are made consistent with NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plant,'' April 2001, and limit unnecessary surveillance testing when the ABVS is actively performing its safety function.

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

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

The proposed changes revise the frequency (from 18 months to 24 months), and acceptance criteria for laboratory testing of the charcoal adsorbers in the engineered safety feature (ESF) ventilation systems. The testing is performed offsite on charcoal samples taken from the ventilation systems, and would have no impact on any accident initiator, or change the consequences of any previously analyzed accident. Continued compliance with industry standards and Diablo Canyon Power Plant test data ensure that the revised requirements would continue to ensure the charcoal adsorbers are capable of performing their intended safety function; therefore, the changes would not affect the accident mitigation capabilities of the ESF ventilation systems.

The preheaters in the control room ventilation system (CRVS) and auxiliary building ventilation system (ABVS) are not initiators of analyzed events, are no longer credited in mitigating design basis accidents or transients, and are therefore not required for system operability. The deletion of the requirement to demonstrate the capability of the preheaters every 24 months, and the changes to the action requirements and surveillance requirements for the CRVS and ABVS would not affect the assumed accident mitigation capabilities of these ESF ventilation systems.

The proposed changes also provide for two trains of the CRVS to be inoperable for up to 24 hours as a result of the CRVS boundary being inoperable. This allowance is contingent on providing and implementing proceduralized compensatory measures to restore the boundary during that time period. Although this change does provide for an increase in the allowed time for continued plant operation in the applicable modes, its acceptability is based on the low probability of any design basis accident during that time period and the protection provided by the compensatory measures that would be established. In addition, this change has no impact on any accident initiator, and does not change the consequences of any previously analyzed accident, because the administrative controls will restore the boundary before it is required to protect control room personnel.

The proposed changes also provide for limiting the applicability of surveillance requirement (SR) 3.7.12.3, which verifies the operability of the ABVS on a safety injection (SI) signal. The limitation is imposed only when the ABVS is aligned and operating in its safety function configuration. Since the ABVS is already performing its safety function when it is in that condition, verifying the automatic capability to transfer to that configuration is unnecessary. Since this limitation is only during periods where the ABVS is in its safety function configuration it has no impact on any accident initiator, or change the consequences of any previously analyzed accident. In addition, this surveillance is still required to be current whenever the ABVS is returned to automatic.

The proposed changes also provide for limiting the required operability of the fuel handling building ventilation system (FHBVS) based on a minimum time period that all fuel assemblies in the fuel pool have not been part of a critical core. This change does reduce the current operability requirements for the FHBVS and increases the consequences of a fuel handling accident with the FHBVS inoperable. However, limiting the FHBVS operability requirements does not increase the probability of any accident, and as determined in the new fuel handling accident (FHA) analysis, the potential release levels are still well within acceptable limits and do not significantly increase the consequences of a FHA.

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

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

The ABVS, FHBVS and CRVS are accident response systems and as such cannot create accidents. The changes to the charcoal sample test requirements will not affect the method of operation of the systems. The proposed changes only affect the laboratory test acceptance criteria for the charcoal samples, and how the charcoal preheaters are credited for meeting technical specification (TS) requirements. These changes result in a more conservative testing methodology. Deletion of the preheater requirements from the TS is based on the heaters not being credited for mitigation of any accident condition and does not affect the operation of these systems. The design and operation of the CRVS, ABVS, and FHBVS are not affected by these changes. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes.

The proposed changes also provide for two trains of the CRVS to be inoperable for up to 24 hours as a result of the CRVS boundary being inoperable. This allowance requires proceduralized compensatory measures to protect the operators during that time period. Although this change does provide for an increase in the allowed time for continued plant operation, its acceptability is based on the low probability of any design basis accident during that time period and the protection provided by the compensatory measures that would be established. The design and operation of the control room ventilation system is not affected by this change.

The proposed changes also provide for limiting the applicability of SR 3.7.12.3, which verifies the operability of the ABVS on an SI signal. The limitation is imposed only when the ABVS is aligned and operating in its safety function configuration. Since the ABVS is already performing its safety function when it is in this condition, verifying the automatic capability to transfer to this configuration is unnecessary. Since this limitation is only during periods where the ABVS is in its safety function configuration, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes also provide for limiting the required operability of the FHBVS based on a minimum time period that all fuel assemblies in the fuel pool have not been part of a critical core. This change does reduce the current operability requirements for the FHBVS by limiting these requirements to the period when the system would be required to mitigate the radiological consequences of an accident to acceptable limits. However, the design and operation of the FHBVS is not affected by this change.

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

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

The charcoal adsorber sample laboratory testing protocol accurately demonstrates the required performance of the adsorbers in the CRVS and ABVS following a design basis accident or in the FHBVS following a fuel handling accident outside containment. The changes in charcoal testing acceptance criteria and frequency will not affect system performance or operation. They will continue to ensure that the charcoal will perform its safety function. The decontamination efficiencies used in the offsite and control room dose analyses are not affected by this change. Therefore the offsite and control room dose analyses are not affected by this change, and offsite and control room doses will remain within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, GDC [General Design Criterion] 19. Although there is a reduction in the safety factor provided by the previous testing protocol, the revised testing protocol follows current industry standards. These standards ensure adequate margin exists and that the charcoal will perform its design basis function. As a result, there is no significant reduction is [in] a margin of safety.

The proposed changes also provide for two trains of the CRVS to be inoperable for up to

24 hours as a result of the control room boundary being inoperable. Although this change does provide for an increase in the allowed time for continued plant operation under certain conditions, its acceptability is based on a low probability of any design basis accident occurring during that time period and the added protection provided by the compensatory measures that would be established. The increase in inoperability could be considered to be a decrease in the margin of safety of this system. However, based on the low probability of a concurrent accident requiring system operability during the completion time for this condition and the ability of the compensatory measures to restore the boundary before it is needed if an accident occurs, this potential reduction in safety margin is not considered to be significant.

The proposed changes also provide for limiting the applicability of SR 3.7.12.3, which verifies the operability of the ABVS on a SI signal. The limitation is imposed only when the ABVS is aligned and operating in its safety function configuration. Since the ABVS is already performing its safety function when it is in this condition, verifying the automatic capability to transfer to this configuration is unnecessary. Since this limitation is only during periods where the ABVS is already in its safety function configuration, the margin of safety is actually increased because the ABVS does not have to change configuration as a result of an accident to perform its safety function.

The proposed changes also provide for limiting the required operability of the FHBVS based on a minimum time period (''recently irradiated fuel'') that all fuel assemblies in the fuel pool have not been part of a critical core. This change does reduce the current operability requirements for the FHBVS by limiting operability to the period when the system would be required to mitigate the radiological consequences of an accident to acceptable limits. This proposed change creates the potential for increased dose in the control room and at the site boundary due to a FHA outside containment. However, the new analysis demonstrates that the resultant doses are well within the Regulatory Guide (RG) 1.183 limits and within the GDC 19 limits. In the case of the offsite dose values, they remain within the RG 1.183 limits, which is considered acceptable. Based on this, the margin of safety is not significantly reduced.

In the new FHA analysis, the offsite and control room doses due to a FHA outside containment have been evaluated using conservative assumptions, such as no credit being taken for the functionality of either FHBVS train's activated charcoal adsorber sections, the control room ventilation system remains in normal mode with no charcoal filtration available, and all airborne activity caused by the FHA is released at a linear rate over two hours. These conservative assumptions ensure the results of the calculation bounds the expected dose. The normal availability of the fuel handling building and control room filtration systems will reduce the potential control room and offsite doses in the event of a FHA, and provides additional margin to the calculated doses.

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

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

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, PO Box 7442, San Francisco, CA 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: May 29, 2003.

Description of amendment requests: The proposed changes to the technical specifications would extend the completion time for restoring an inoperable diesel generator from 7 days to 14 days.

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

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

The proposed changes revise the Technical Specification (TS) 3.8.1 completion times for Required Actions A.2 and B.4 associated with the diesel generators (DGs). The proposed changes allow an extension of the current TS completion time from 7 days to 14 days for an inoperable DG.

The proposed changes do not affect the design of the DGs, the operational characteristics or function of the DGs, the interfaces between the DGs and other plant systems, or the reliability of the DGs. Required Actions and the associated completion times are not initiating conditions for any accident previously evaluated, and the DGs are not initiators of any previously evaluated accidents. The DGs mitigate the consequences of previously evaluated accidents including loss of offsite power. The consequences of a previously analyzed event will not be significantly affected by the extended DG completion time since the DGs will continue to be capable of performing their accident mitigation function as assumed in the accident analysis. Thus the consequences of accidents previously analyzed are unchanged between the existing TS requirements and the proposed changes. The consequences of an accident are independent of the time the DGs are out of service as long as adequate DG availability is assured. The proposed changes will not result in a significant decrease in DG availability so that the assumptions regarding DG availability are not impacted.

To fully evaluate the effect of the proposed DG completion time extension, probabilistic risk assessment methods and a deterministic analysis were utilized. The results of the analysis show no significant increase in core damage frequency and large early release frequency.

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 accident from any accident previously evaluated.

The proposed changes do not involve a change in the design, configuration, or method of operation of the plant. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure that the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

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

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

The proposed 14 day DG completion time is based upon both a deterministic evaluation and a risk-informed assessment. The availability of offsite power coupled with the availability of the other DGs in the affected unit, the unit auxiliary feedwater pumps, and all auxiliary saltwater trains (including the cross-tie) and utilization of the Online Risk Management Program while a DG is inoperable, provide adequate compensation for the potential small incremental increase in plant risk of the extended DG completion time. In addition, the increased availability of the DGs during refueling outages provides a reduction in plant risk during shutdown periods.

The risk assessment performed to support this license amendment request concluded that the increase in plant risk is small and consistent with the NRC's Safety Goal Policy Statement, ''Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,'' **Federal Register**, Volume 60, p. 42622, August 16, 1995 and guidance contained in [* * *] Regulatory Guides (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998 and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications,'' dated August 1998. Together, the deterministic evaluation and the risk-informed assessment provide high assurance of the capability to provide power to the engineered safety feature buses during the proposed 14 day DG completion time.

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

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

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 6, 2003.

Description of amendment request: The proposed amendments would delete SSES 1 and 2 Technical Specifications (TSs) 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' and revise TS 3.4.1, ''Recirculation Loops Operating.'' These changes would reverse approved TS Amendment Nos. 184 (Unit 1) and 158 (Unit 2) dated July 30, 1999, that are not yet implemented, which effectively results in no change to the current SSES 1 and 2 operation. Extension of the implementation date was needed to provide time to address continuing hardware and software deficiencies with the OPRM system. The extension of the implementation date until November 1, 2001, was approved by Amendment Nos. 187 (Unit 1) and 161 (Unit 2) dated June 2, 2000. A second extension of the implementation date until November 1, 2003, was approved by Amendment Nos. 196 (Unit 1) and 172 (Unit 2) dated October 29, 2001. This deferral was based on a Title 10 Code of Federal Regulations (10 CFR), part 21, report issued by General Electric Company on August 31, 2001, which identified a non-conservative deficiency in the OPRM trip setpoint methodology. The licensee stated that the OPRM system cannot be declared OPERABLE until a revised NRC-approved methodology providing a valid basis for the trip setpoints is available and adopted for the SSES 1 and 2 OPRM systems. The implementation requirements associated with Amendment Nos. 187, 161, 196 and 172 would also be superceded with this proposed amendment. The proposed amendment would formally reinstate the requirements currently governing operation, which define appropriately conservative restrictions to plant operation and operator response to thermal hydraulic instability events.

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

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

Response: No.

The OPRM system is not an initiator to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). The changes do not involve a physical change to structures, systems, or components (SSCs) since the RPS [reactor protection system] trip function has not been installed and does not alter the method of operation or control of SSCs since the OPRM system has not been declared OPERABLE. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents (including assumed protection of fuel design limits) are unaffected by these changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed Technical Specification (TS) ensures that the protection from thermal hydraulic instabilities remains as previously evaluated and the protection for fuel design limits remain as described in the FSAR. Therefore, the mitigative functions will continue to provide the protection assumed by the existing analysis.

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

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

Response: No.

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

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change is acceptable because the required protection from thermal hydraulic instabilities remains as previously evaluated and the protection for fuel design limits remain as described in the FSAR. Operation in accordance with the proposed TS ensures that the margin of safety is maintained. Therefore, the 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.

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

Date of amendment request: May 14, 2003.

Description of amendment request: The proposed amendment revises surveillance requirement 4.6.2.1 for demonstrating operability of containment spray system spray nozzles.

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

The Containment Spray System is not considered an initiator of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that may initiate an analyzed event. The proposed change will not alter the operation or otherwise increase the failure probability of any plant equipment that can initiate an analyzed accident.

This change does not affect the plant design. There is no increase in the likelihood of formation of significant corrosion products. Due to their location at the top of the containment, introduction of foreign material into the spray headers is unlikely. Foreign material introduced during maintenance activities would be the most likely source for obstruction, and verification following such maintenance would confirm the nozzles remain unobstructed.

Consequently, there is no significant increase in the probability of an accident previously evaluated.

The Containment Spray System is designed to address the consequences of a LOCA [loss of coolant accident]. The Containment Spray System is capable of performing its function effectively with the single failure of any active component in the system, any of its subsystems, or any of its support systems. A plugged nozzle would have negligible impact on the capability of the Containment Spray System to respond to a Loss of Coolant Accident.

Therefore, the consequences of an accident previously evaluated are not significantly affected by the proposed change.

2. Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change will not physically alter the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The system is not susceptible to corrosion-induced obstruction or obstruction from sources external to the system. Maintenance activities that could introduce foreign material into the system would require subsequent verification to ensure there is no nozzle blockage. The spray header nozzles are expected to remain unblocked and available in the event that the safety function is required. Therefore, the capacity of the system would remain unaffected. Hence, this 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 22, 2003.

Description of amendment request: The proposed amendments revise Technical Specification (TS) 3.4.2.2, ''Reactor Coolant System,'' to relax the lift setting tolerance of the pressurizer safety valves from ±2 percent to +2 percent, -3 percent.

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

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

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from power event already evaluated in the UFSAR [Updated Final Safety Analysis Report]. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from [the] power event already evaluated in the UFSAR. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (*i.e.*, fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from power event already evaluated in the UFSAR. That analysis demonstrated that the fuel design limits were maintained by the reactor protection system since the DNBR [Departure from Nucleate Boiling Ratio] was maintained above the limit value. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92 and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 5, 2003 (TSC 03–08).

Description of amendment request: The proposed amendment would revise the reactor coolant system (RCS) heatup and cooldown curves (pressure-temperature (P–T) limits). The revision replaces the P–T limits that are currently analyzed for 14.5 Effective Full Power Years (EFPYs) with new limits analyzed for 32 EFPYs. In addition, the amendment includes corresponding changes to the Technical Specification (TS) figure associated with the Low Temperature Over Pressure Protection and the TS Bases.

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

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

No. The proposed revision does not affect plant equipment, test methods or operating practices. The modification to SQN TSs is consistent 10 CFR 50, Appendix G in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1.'' The proposed change continues to provide controls for safe operation within the required limits. The proposed changes do not contribute to events or assumptions associated with postulated design basis accidents (DBA). The proposed revisions continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

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

No. The proposed revision is not the result of changes to plant equipment, test methods, or operating practices. The proposed revision to the SQN Unit 2 P–T limits continues to ensure that conservative fracture toughness margins are maintained to protect against reactor pressure vessel failure. In addition, SQN's current setpoints for low-temperature overpressure protection were evaluated and are bounding for the proposed 32 EFPY P–T limits. The updated P–T limits are based on NRC approved methodology in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1.''

The reactor vessel P–T limits are operational limits and are not considered to be contributors to the generation of postulated accidents. The safety functions of the associated systems remain unchanged and do not affect the assumptions of DBAs. The operational limits continue to be governed within the TSs. Accordingly, the proposed change does not create the possibility of a new or different kind of accident.

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

No. TVA's proposed TS amendment provides revised reactor pressure vessel P–T limits that are within the design capabilities of the pressure control systems for protection of the RCS. The limits are based on conservative design margins that ensure that plant operation is within the design capacity of the reactor vessel materials. Accordingly, the function of the RCS to provide a fission product barrier is not compromised.

TVA's proposed change to revise P–T limits does not result in a change to system design features. The proposed change does not affect plant conditions that result in precursors to accidents or cause degradation of accident mitigation systems. The plant system safety functions are not altered by the proposed change.

The proposed changes allow plant operation with different P–T limits while continuing to retain conservative margins for assuring integrity of the reactor vessel and the RCS. Consequently, the proposed TS revisions do not significantly reduce 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, TN 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: June 5, 2003 (TSC 03–09).

Description of amendment request: The proposed change to the Updated Final Safety Analysis Report (UFSAR) would amend the design and licensing basis to identify that operator action may be necessary to ensure containment design pressure is not exceeded subsequent to a high energy line break (HELB) such as loss-of-coolant-accident.

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 procedure changes/additions being implemented to mitigate a SCSA [station control and service air] leak in containment will only be used following a HELB in containment, a consequential rupture of an SCSA line and a failure of the outboard CIV [containment isolation valve] on the SCSA containment supply.

Operators isolate the SCSA leak on the accident unit by either manually closing a valve upstream of the stuck-open CIV or by shutting down the station air compressors. If the station air compressors are shut down prior to performing an emergency shut down of the non-accident unit or if an operator error results in an isolation of the control air supply to the non-accident unit, then at-worst, a UFSAR Condition II event is induced on the non-accident unit. For example a reactor trip from full power or a loss of normal feedwater—loss of control air to the feedwater regulator valves resulting in a loss of normal feedwater to the non-accident unit.

A UFSAR Condition II event has a frequency of one per year. Therefore, the proposed procedure changes/additions, including the potential for operator error do not result in more than a minimal increase in a previously evaluated Condition II event (1+1/40 = 1.025 less than 10 percent increase).

The operator actions being implemented to mitigate a SCSA leak in containment are performed after the occurrence of an accident on primarily non-safety-related systems, structures or components [SSCs] so they do not increase the likelihood of the occurrence of a malfunction of equipment previously evaluated in the UFSAR.

The air operated containment isolation valve is assumed to fail open due to single failure criteria and, containment isolation/integrity is maintained by the inboard check valve. The containment boundary is unaffected by the operator actions being implemented to mitigate a SCSA leak in containment. Therefore, the consequences of all accidents previously evaluated in the UFSAR remain unchanged.

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

No. This change implements new manual actions for accident failure modes not previously evaluated in the UFSAR. The manual actions are required to ensure containment design pressure is not exceeded.

Operators isolate the SCSA leak on the accident unit by either manually closing an upstream isolation valve on the accident unit's SCSA containment supply or by shutting down the station air compressors. The operator actions being implemented have been determined to meet the criteria for safety-related operator actions in NRC Information Notice (IN) 97–78/ANS–58.8 and; therefore, there are no credible operator actions which would prevent isolation of a SCSA leak prior to containment design pressure being exceeded. If the station air compressors are shut down prior to performing an emergency shutdown of the non-accident unit or if an operator error results in an isolation of the SCSA supply to the non-accident unit, then at-worst, a UFSAR Condition II event occurs. Because UFSAR Condition II events have been previously identified, the operator actions being added under this change do not create the possibility of an accident of a different type than previously evaluated.

The operator actions being implemented to mitigate a SCSA leak in containment are performed after the occurrence of an accident on primarily non-safety-related SSCs so they do not create a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the UFSAR.

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

No. The established limits for the fuel, reactor vessel or containment are not affected by the addition of operator actions to isolate a SCSA leak inside containment. Isolation of the air leak within two hours of a large break loss-of-coolant accident (LBLOCA) prevents containment pressure exceeding the peak calculated pressure. Consequently, this change does not represent a reduction in the margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: May 14, 2003.

Description of amendment request: The proposed amendment would allow an alternate Westinghouse methodology for the measurement of reactor coolant system (RCS) total flow rate via measurement of the RCS elbow tap differential pressures. TVA stated that this methodology is similar to that reviewed and approved by the NRC for other utilities.

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. TVA's evaluation for WBN Unit 1 determined that the probability of an accident will not increase since adequate RCS flow will still be assured. Sufficient margin exists to account for all reasonable instrument uncertainties; therefore, no changes to installed equipment or hardware in the plant are required, thus the probability of an accident occurring remains unchanged. The initial conditions for all accident scenarios modeled are the same and the conditions at the time of trip, as modeled in the various safety analyses are the same. Therefore, the consequences of an accident will be the same as those previously analyzed.

Therefore, since the actual plant configuration, performance of systems, and initiating event mechanisms are not being changed, TVA has concluded that 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. There are no changes in operation of the plant that could introduce a new failure mode. No new accident scenarios have been identified. Operation of the plant will be consistent with that previously modeled, *i.e.*, the time of reactor trip in the various safety analyses is the same, thus plant response will be the same and will not introduce any different accident scenarios that have not been evaluated.

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

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

No. The proposed change reflects changes due to the method used to verify RCS flow at the beginning of each cycle. However, no changes to the Safety Analysis assumptions were required; therefore, the margin of safety will remain the same. Therefore, TVA concludes that the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: December 16, 2002, as supplemented on April 1, 2003.

Brief description of amendment: The amendment revised the Technical Specifications, changing the safety limit minimum critical power ratio (SLMCPR) from 1.11 to 1.09 for both four- or five-recirculation-loop operation, and from 1.12 to 1.10 for three-recirculation-loop operation. It also added a paragraph to explain that the lower SLMCPR values are due primarily to an improved treatment of the power distribution uncertainty.

Date of issuance: June 5, 2003.

Effective date: June 5, 2003 and shall be implemented within 30 days of issuance.

Amendment No.: 238.

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

*Date of initial notice in **Federal Register:*** January 21, 2003 (68 FR 2799).

The April 1, 2003, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 5, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 2, 2002, as supplemented by letter dated April 14, 2003.

Brief description of amendments: The amendments revise the Technical Specifications for Administrative Controls in Section 5.0 concerning Responsibility, Unit Staff, Unit Staff Qualifications, and controls for the High Radiation Area.

Date of issuance: June 6, 2003.

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

Amendment Nos.: 213 and 194.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 7, 2003 (68 FR 800).

The supplement dated April 14, 2003, provided clarifying information that did not change the scope of the December 2, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 30, 2002, and its supplement dated April 28, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 2.1.1.2, ''Minimum Critical Power Ratio Safety Limit

(MCPRSL)'' to support operating during Cycle 17. Cycle 17 is the first cycle of operation with a mixed core of ABB/CE/ Westinghouse SVEA–96 fuel and Framatome ANP Atrium™-10 reload fuel. The amendment also revises Surveillance Requirement (SR) 3.3.1.3.2—the low power range monitor (LPRM) calibration frequency specified in the TS for the oscillation power range monitor. This change corrects an inconsistency between the LPRM calibration frequency specified in SR 3.3.1.3.2 and SR 3.3.1.1.7, ''Reactor Protection System (RPS) Instrumentation.''

Date of issuance: June 2, 2003.

Effective date: June 2, 2003, and shall be implemented before the plant restarts after completion of Refueling Outage 16.

Amendment No.: 186.

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

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

The April 28, 2003, supplemental letter provided additional clarifying information, did not change the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 2, 2003.

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: October 4, 2002.

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

Date of issuance: June 2, 2003.

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

Amendment No.: 215.

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

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68736).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 2, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 23, 2002, as supplemented January 24 and April 21, 2003.

Description of amendment request: The amendment relocates, intact, Technical Specification (TS) 6.2.3, ''Independent Technical Reviews;'' TS 6.4, ''Review and Audit;'' TS 6.7.2 through 6.7.5 (specific descriptions of the procedure review and approval process); and TS 6.9, ''Records Retention'' to the Operational Quality Assurance Program. The amendment also changes the title of the senior onsite official from ''Executive Vice President and Chief Nuclear Officer'' to ''Site Vice President,'' revises the 10 CFR 20 references in the TSs to bring them into consistency with 10 CFR 20, and makes other minor editorial changes.

Date of issuance: June 6, 2003.

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

Amendment No.: 88.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 18, 2003 (68 FR 7817).

The April 21, 2003, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expanded the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 6, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 10, 2002, as supplemented by letter dated April 16, 2003.

Brief description of amendment: The amendment replaces the fire protection requirements contained in Facility Operating License (FOL) Section 2.C.(4) with the standard fire protection FOL condition recommended by Generic Letter 86–10, Section F, adapted to Cooper Nuclear Station.

Date of issuance: June 5, 2003.

Effective date: June 5, 2003.

Amendment No.: 199.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 7, 2003 (68 FR 808).

The supplement dated April 16, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 5, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 7, 2002, as supplemented by letter dated March 24, 2003.

Brief description of amendment: The amendment (1) adds a new Surveillance Requirement (SR) 4.0.3 to extend the delay period, up to 24 hours or up to the limit of the specified frequency, whichever is greater, before entering a Limiting Condition for Operation following a missed surveillance; (2) adds a new SR 4.0.1 to define general conditions for use of SRs; and (3) makes various editorial and administrative changes.

Date of issuance: June 3, 2003.

Effective date: June 3, 2003, to be implemented within 60 days of issuance.

Amendment No.: 182.

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

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68739) and April 29, 2003 (68 FR 22748).

The supplement expanded the scope of the application, and was addressed by the second notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 3, 2003.

No significant hazards consideration comments received: No.

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 application for amendments: June 11, 2002.

Brief description of amendments: These amendments revise Technical Specification 3.1.8, ''Physics Tests Exceptions—Mode 2,'' to correct an error in the numbering of a function. Specifically, the reference in Limiting Condition for Operation 3.1.8 to Function 17.d has been changed to Function 17.e.

Date of issuance: June 3, 2003.

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

Amendment Nos.: 208 & 213.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56325).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 6, 2003.

Brief description of amendments: The amendments revise Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 of the Technical Specifications on the reactor trip system instrumentation. The proposed changes to SR 3.3.1.2 move Note 1 to the body of the SR, replace the reference to nuclear instrumentation system channel output by a reference to power range channel output, and delete the reference to the absolute difference. The change to SR 3.3.1.3 is editorial.

Date of issuance: June 2, 2003.

Effective date: June 2, 2003, and shall be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1–157; Unit 2–157.

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

*Date of initial notice in **Federal Register:*** April 15, 2003 (68 FR 18282).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 3, 2003, and its supplement dated March 5, 2003.

Brief description of amendments: The amendment authorizes revisions to the Final Safety Analysis Report (FSAR) Update to incorporate the NRC approval of a probability of detection of 1.0 to one bobbin indication, contained in Diablo Canyon Nuclear Power Plant (DCPP) Unit 2 steam generator 4 tube at row 44, column 45 at the second tube support plate on the hot leg side, for the beginning of cycle voltage distribution for the DCPP Unit 2 Cycle 12 operational assessment. In a **Federal Register** notice dated April 15, 2003 (68 FR 18284), the NRC described the amendment request as follows:

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

The change from a TS to an FSAR revision resulted from the March 5, 2003, supplement and is not substantial in that the technical issues and no significant hazards consideration determination remain the same.

Date of issuance: June 3, 2003.

Effective date: June 3, 2003, and shall be implemented within 30 days of the date of issuance. The implementation of the amendment includes the incorporation into the FSAR Update the changes discussed above, as described in the licensee's application dated March 3, 2003, its supplement dated March 5, 2003, and evaluated in the staff's safety evaluation attached to the amendment.

Amendment No.: 158.

Facility Operating License No. DPR–82: The amendment authorized revision of the FSAR Update.

*Date of initial notice in **Federal Register:*** April 15, 2003 (68 FR 18284).

The March 5, 2003, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 31, 2002.

Brief description of amendments: These amendments revised the Technical Specifications, Section 3.7.6, ''Main Turbine Bypass System,'' to change the requirement for operability of the main turbine bypass system bypass valves. Specifically, Surveillance Requirement 3.7.6 would be revised to test only each required turbine bypass valve every 31 days.

Date of issuance: May 29, 2003.

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

Amendment Nos.: 210 and 185.

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

*Date of initial notice in **Federal Register:*** December 24, 2002 (67 FR 78524).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 29, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 30, 2002.

Brief description of amendments: These amendments revise Technical Specifications Section 5.5.7, ''Ventilation Filter Testing Program,'' to change the control room emergency outside air supply system (CREOASS) maximum allowed filter train pressure drop from <9.1 inches water gage (wg) to <7.3 inches wg.

Date of issuance: May 29, 2003.

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

Amendment Nos.: 211 and 186.

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

*Date of initial notice in **Federal Register:*** December 24, 2002 (67 FR 78523).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 29, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 3, 2003.

Brief description of amendments: The amendments delete Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminate the requirements to have and maintain the post-accident sampling systems. The amendments also address related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: June 3, 2003.

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

Amendment Nos.: 212 and 187.

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

*Date of initial notice in **Federal Register:*** April 29, 2003. (68 FR 22752).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 31, 2002.

Brief description of amendments: These amendments revised Technical Specifications, Section 3.3.6.1, ''Primary Containment Isolation Instrumentation,'' to add an ACTIONS Note allowing intermittent opening, under administrative control, of penetration flow paths that are isolated. Additionally, these amendments revised TSs Section 3.3.6.1 to breakout the traversing incore probe system isolation as a separate isolation Function with an associated Required Action to isolate the penetration within 24 hours rather than immediately initiating a unit shutdown.

Date of issuance: June 5, 2003.

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

Amendment Nos.: 213 and 188.

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

*Date of initial notice in **Federal Register:*** December 24, 2002 (67 FR 78523).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 5, 2003.

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: May 8, 2002, as supplemented by letters dated November 26, 2002, and April 10, 2003.

Brief description of amendments: The amendments revise the Reactor Core Safety Limits curve in Technical Specifications (TS) Figure 2.1.1–1, and the Over Temperature Delta Temperature (OTDT) and Over Power Delta Temperature (OPDT) reactor trip functions described in TS Table 3.3.1–1. These changes will provide Vogtle Electric Generating Plant (VEGP), Units 1 and 2 with increased operating margins that will increase the OTDT and OPDT setpoints to account for hot leg temperature fluctuations that are part of the VEGP Setpoint Margin Recovery Program.

Date of issuance: June 4, 2003.

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

Amendment Nos.: 128/106.

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

Date of initial notice in **Federal Register:** July 23, 2002.

The supplements dated November 26, 2002, and April 10, 2003, provided clarifying information that did not change the scope of the May 8, 2002, application nor the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 4, 2003.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Units 1and 2, Hamilton County, Tennessee

Date of application for amendment: October 4, 2002, as supplemented February 19, 2003, and May 19, 2003.

Brief description of amendment: The amendments revise Technical Specification (TS) 6.8.4.h, Containment Leakage Rate Testing Program, to allow the licensee to postpone its Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT) for 5 years. Specifically, for Unit 1 the performance of the spring 2003 ILRT may be deferred up to 5 years but no later than spring 2008, and for Unit 2 performance of the fall 2003 ILRT may be deferred up to an additional 3.5 years but no later than spring 2007. In Amendment No. 265 to the Facility Operating License No. DPR–79 for SQN, Unit 2, TS 6.8.4.h was revised to allow the licensee to postpone the ILRT one cycle (*i.e.*, 1.5 years) from spring 2002. Therefore, the total deferral for SQN, Unit 2 from the original requirement to perform a ILRT in spring 2002 will be up to 5 years.

Date of issuance: May 29, 2003.

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

Amendment Nos.: 287 and 276.

Facility Operating License No. DPR–77: Amendment revises the technical specifications.

Date of initial notice in **Federal Register:** February 4, 2003 (68 FR 5681).

The February 19, and May 19, 2003, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 29, 2003.

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: November 19, 2002, as supplemented by letters dated February 5 and May 5, 2003.

Brief description of amendments: The amendments revise Appendix B to the Facility Operating License, Environmental Protection Plan (EPP), to replace references to the U.S. Environmental Protection Agency's National Pollutant Discharge Elimination System expired permit. The amendments also contain minor changes to the EPP to be consistent with the provisions of the current Texas Pollutant Discharge Elimination System permit and the Final Environmental Statement—Operating License Stage, and consolidate the Unit 1 and Unit 2 EPPs into a single document.

Date of issuance: May 29, 2003.

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

Amendment Nos.: 104 and 104.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Facility Operating License, Appendix B, ''Environmental Protection Plan.''

Date of initial notice in **Federal Register:** December 24, 2002 (67 FR 78524).

The 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 May 29, 2003.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: March 21, 2003.

Brief description of amendment: The amendment revises paragraphs in Section 5.0, ''Administrative Controls,'' of the Technical Specifications to allow the use of generic personnel titles in place of plant-specific personnel titles and requires either the operations manager or assistant operations manager to hold a senior reactor operator license.

Date of issuance: June 3, 2003.

Effective date: June 3, 2003, and shall be implemented within 30 days of the date of issuance, including the incorporation of the Final Safety Analysis Report changes described in the licensee's application dated March 21, 2003, and the staff's Safety Evaluation for this amendment.

Amendment No.: 155.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 16, 2003 (68 FR 18714).

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

No significant hazards consideration comments received: No.

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By July 24, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

Date of application for amendments: May 1, 2003, as supplemented May 2 and May 15, 2003.

Brief description of amendments: The amendments modify technical specification surveillance requirements to provide an alternative means of testing the Unit 1 main steam Electromatic relief valves, including those that provide the automatic depressurization and the low set relief functions, and provide an alternative means for testing the Units 1 and 2 dual function Target Rock safety/relief valves.

Date of issuance: May 28, 2003.

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

Amendment Nos.: 216/210.

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. 68 FR 25645, dated May 13, 2003. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The supplements dated May 2 and May 15, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed NSHC determination. 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 May 28, 2003.

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

NRC Section Chief: Anthony J. Mendiola.

Dated at Rockville, Maryland, this 16th day of June 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–15597 Filed 6–23–03; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement Regarding Extension of Reactor Coolant Pump Motor Flywheel Examination for Westinghouse Plants Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for comment.

SUMMARY: Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model safety evaluation (SE) relating to a change in the technical specification