a public scoping meeting to help identify significant issues related to a proposed activity and to determine the scope of issues to be addressed in an EIS. The NRC will hold a public meeting for the EIS regarding the Clinton ESP application and the associated site redress plan. The scoping meeting will be held in the Vespasian Warner Public Library, located at 310 N. Quincy Street, Clinton, Illinois, on Thursday, December 18, 2003. The meeting will convene at 7 p.m. and will continue until 9:30 p.m., as necessary. The meeting will be transcribed and will include the following: (1) An overview by the NRC staff of the NEPA environmental review process, the proposed scope of the EIS, and the proposed review schedule; and (2) the opportunity for interested Government agencies, organizations, and individuals to submit comments or suggestions on the environmental issues or the proposed scope of the EIS. Additionally, the NRC staff will host informal discussions one hour prior to the start of the meeting at the Vespasian Warner Public Library. No formal comments on the proposed scope of the EIS will be accepted during the informal discussions. To be considered, comments must be provided either during the transcribed portion of the public meeting or in writing, as discussed below. Persons may pre-register to attend or present oral comments at the meeting on the scope of the NEPA review by contacting Ms. Jennifer Davis by telephone at 1 (800) 368–5642, extension 3835, or by Internet at *ClintonEIS@nrc.gov* no later than December 5, 2003. Members of the public may also register to speak at the meeting within 15 minutes of the start of the session. Individual oral comments may be limited by the time available, depending on the number of persons who register. Members of the public who have not registered may also have an opportunity to speak, if time permits. Public comments will be considered in the scoping process for the EIS. If special equipment or accommodations are needed to attend or present information at the public meeting, the need should be brought to Ms. Davis' attention no later than December 5, 2003, so that the NRC staff can determine whether the request can be accommodated.

Members of the public may send written comments on the environmental scope of the Clinton ESP and site redress plan review to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Mailstop T–6D59, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Comments may also be hand-delivered to the NRC at 11545 Rockville Pike, Rockville, Maryland, Room T–6D59, from 7:30 a.m. to 4:15 p.m. during Federal workdays. To be considered in the scoping process, written comments should be postmarked by January 9, 2004. Electronic comments may be sent by the Internet at *ClintonEIS@nrc.gov.* Electronic submissions should be sent no later than January 9, 2004, to be considered in the scoping process. Comments will be available electronically and accessible through the NRC's PERR link *http://www.nrc.gov/reading-rm/adams.html* at the NRC Homepage.

Participation in the scoping process for the EIS does not entitle participants to become parties to the proceeding to which the EIS relates. Notice of a hearing regarding the application for an ESP will be the subject of a future **Federal Register** notice.

At the conclusion of the scoping process, the NRC will prepare a concise summary of the determination and conclusions reached, including the significant issues identified, and will send a copy of the summary to each participant in the scoping process. The summary will also be available for inspection through the NRC's PERR link. The staff will then prepare and issue for comment the draft EIS, which will be the subject of separate notices and a separate public meeting. Copies will be available for public inspection at the above-mentioned addresses, and one copy per request will be provided free of charge. After receipt and consideration of the comments, the NRC will prepare a final EIS, which will also be available for public inspection.

Information about the proposed action, the EIS, and the scoping process may be obtained from Ms. Davis at the aforementioned telephone number or e-mail address.

Dated at Rockville, Maryland, this 19th day of November 2003.

For the Nuclear Regulatory Commission.

K. Steven West,

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

[FR Doc. 03–29351 Filed 11–24–03; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, October 31, through November 13, 2003. The last biweekly notice was published on November 12, 2003 (68 FR 64133).

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 December 26, 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: October 14, 2003.

Description of amendments request: The proposed amendment would change the frequency of surveillance testing for some engineered safety features (ESF) components.

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.

Integrated testing of the ESF trains takes place while the unit is shut down. The equipment being tested is normally used to respond to an accident when the Unit is in Modes 1, 2, or 3. Changing the test Frequency to a longer period does not affect the scope of the testing or the methods used during the testing. Therefore, there is no increase in the probability of an accident previously evaluated caused by the testing itself.

The components tested during the integrated ESF test are components needed to mitigate the consequences of an accident. Increasing the length of time between integrated tests increases the likelihood of undetected equipment failure. This creates a change in plant risk. This change in risk is analyzed and quantified using probabilistic risk assessment techniques. The risk analysis provides results that show the proposed increase in ESF component surveillance testing Frequency meets the guidance of Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.'' The increase in risk is within the guidelines of the regulatory guidance. There is no significant change in the probability that the equipment will suffer an undetected failure in the increased time between Surveillance tests. Therefore, there is no significant increase in the consequences o[f] an accident previously evaluated.

An additional change is proposed to delete a Surveillance Requirement because the signal tested in the Surveillance Requirement is no longer installed in the plant. This deletion has no impact on plant operations or the response of the plant in an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequence 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 change would extend the Surveillance Frequency of the integrated ESF test. This change does not affect the scope of the testing or the methods used during the testing. Plant equipment will continue to operate as designed. Only the testing frequency is changed. Because there are no changes in the scope or method of testing and this proposed change does not affect the operation of the equipment in other circumstances, no new accident initiators have been introduced.

An additional change is proposed to delete a Surveillance Requirement because the signal tested in the Surveillance Requirement is no longer installed in the plant. This deletion has no impact on plant operations or the response of the plant and therefore would not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

Surveillance testing is performed to evaluate the operability of equipment used to perform safety functions at the Unit. The components tested during the integrated ESF test are components needed to mitigate the consequences of an accident. Increasing the length of time between integrated tests increases the likelihood of undetected equipment failure. This creates a change in plant risk. This change in risk is analyzed and quantified using probabilistic risk assessment techniques. The risk analysis provides results that show the proposed increase in ESF component surveillance testing Frequency meets the guidance of Regulatory Guide 1.174. The increase in risk is within the guidelines of the regulatory guidance. There is no significant change in the probability that the equipment will suffer an undetected failure in the increased time between Surveillance tests. Since the function of Surveillance testing is to evaluate the operability of equipment, and the increased time between Surveillance tests has been evaluated and found to be acceptable under regulatory guidance, the proposed change would not involve a significant reduction in [a] margin of safety.

An additional change is proposed to delete a Surveillance Requirement because the signal tested in the Surveillance Requirement is no longer installed in the plant. This deletion has no impact on plant operations or the response of the plant in an accident and does not impact the margin of safety.

Therefore, this proposed change does not significantly reduce [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 amendments request involves no significant hazards consideration.

Attorney for licensee: James M. Petro, Jr., Esquire, Counsel, Constellation Energy Group, Inc., 750 East Pratt Street, 5th floor, Baltimore, MD 21202.

NRC Section Chief: Richard J. Laufer.

Consumers Energy Company, Docket No. 50–155, Big Rock Point Nuclear Plant, Charlevoix County, Michigan.

Date of amendment requests: August 6, 2003.

Description of amendment requests: The Big Rock Point Plant is in the 6th year of decommissioning. The reactor was defueled and certified as permanently shutdown by letter to the Nuclear Regulatory Commission dated September 22, 1997. As of March 26, 2003, all the spent fuel has been permanently removed from the plant's spent fuel pool and located to an Independent Spent Fuel Storage Installation (ISFSI). The spent fuel has been loaded into an NRC approved and licensed Spent Fuel Dry Storage System and will be temporarily stored at this installation until such time that a permanent repository is available. The requirements associated with the wet storage of the spent fuel as described in Defueled Technical Specifications are no longer applicable and are being revised.

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

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

No. The proposed change is an administrative change to update the facility's Operating License and Defueled Technical Specifications to reflect the permanent removal of the spent fuel from the Spent Fuel Pool. Requirements for safe storage and handling of irradiated fuel, definitions, design features and administrative controls that were applicable to the facility when spent fuel was stored in the spent fuel pool are no longer valid and are being removed to provide clarity to the licensing basis of the facility in its current configuration. The accidents previously evaluated in the Updated Final Hazards Safety Analysis are based on spent nuclear fuel being stored in the spent fuel pool. Since the spent fuel has been permanently removed from the spent fuel pool, the accidents previously analyzed are no longer credible. The spent fuel has been loaded into an NRC approved and licensed Spent Fuel Dry Storage System and will be temporarily stored at this installation until such time that a permanent repository is available. The spent fuel is now controlled by a different set of approved technical specifications issued and approved pursuant to 10 CFR part 72. Therefore, the proposed administrative change does not affect the consequences of any accident described and evaluated in the Updated Final Hazards Summary Report, and the accidents and transients associated with spent fuel stored in the facility's spent fuel pool are no longer applicable.

Therefore, the proposed administrative change to the Operating License and Defueled Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The spent fuel has been loaded into an NRC approved and licensed Spent Fuel Dry Storage System and will be temporarily stored at this installation until such time that a permanent repository is available. In accordance with 10 CFR part 72, ''Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,'' credible accidents have been evaluated as part of the licensing and approval process for the Dry Fuel Storage System. The requirement to evaluate credible accidents has not changed.

Therefore this proposed administrative change does not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed activity is an administrative change to the Operating License and Defueled Technical Specifications to reflect the permanent removal of the spent fuel from the spent fuel pool and does not involve any significant reduction in any margin of safety that is usually associated with the design and performance of systems, structures and components. Requirements for safe storage and handling of irradiated fuel, definitions, design features and administrative controls that were applicable to the facility when spent fuel was stored in the spent fuel pool are no longer applicable and are being removed to provide clarity to the licensing basis of the facility in its current configuration.

Therefore, the proposed administrative 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: David A. Mikelonis, Esquire, Consumers Energy Company, One Energy Plaza, Jackson, MI 49201–2276.

NRC Section Chief: Claudia Craig.

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

Date of amendment request: October 10, 2003.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' Surveillance Requirement (SR) 3.7.3.6, to permit a one-time extension of SR 3.7.3.6 until startup from the next refueling outage (RF–10) to preclude a mid-cycle shutdown solely for the performance of this SR. SR 3.7.3.6 requires verifying that unfiltered inleakage from CREF system duct work outside the control room envelope that is at negative pressure during accident conditions is within limits. This SR is required to be performed every 36 months, and can be performed only when the CREF system is not required to be Operable (*i.e.*, in MODES 4 or 5, with no operations with a potential for draining the reactor vessel and with no fuel movement of recently irradiated fuel in progress).

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

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

The proposed change allows a one-time extension of SR 3.7.3.6 until startup from the next refueling outage (approximately 10 to 12 months beyond its critical completion date). The Control Room Emergency Filtration (CREF) system provides a configuration for mitigating radiological consequences of accidents; however, it is not considered an initiator of any previously analyzed accident. Therefore, the proposed change cannot increase the probability of any previously evaluated accident.

The CREF system provides a radiologically controlled environment from which the plant can be safely operated following a radiological accident. The current TS surveillance (SR 3.7.3.6) measures inleakage from four sections of CREF system duct work outside the Control Room Envelope (CRE) that are at negative pressure during accident conditions. Based on the results of previous surveillance testing, and the continued performance of SR 3.7.3.3 and 3.7.3.5 on their normal schedule, the delay in performing SR 3.7.3.6 by approximately 10 to 12 months will provide essentially the same degree of assurance that CRE integrity is being maintained as before. It is expected that CRE integrity will remain essentially unchanged from what it is today. Therefore, the proposed change does not significantly increase the radiological consequences of any previously analyzed accident.

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

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

The proposed change to allow a one-time extension of SR 3.7.3.6 until startup from the next refueling outage (approximately 10 to 12 months beyond its critical completion date) does not alter the design or function of the system involved, nor does it introduce any new modes of plant or CREF system operation. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The proposed change to allow a one-time extension of SR 3.7.3.6 until startup from the next refueling outage (approximately 10 to 12 months beyond its critical completion date) will not affect the radiological release from a design basis accident. Based on the results of previous surveillance testing and the continued performance of SR 3.7.3.3 and 3.7.3.5 on their normal schedule, the delay in performing SR 3.7.3.6 by approximately 10 to 12 months will provide essentially the same degree of assurance that CRE integrity is being maintained as existed before; and, the postulated dose to the control room occupants as a result of an accident will remain approximately the same. Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi.

Date of amendment request: October 24, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to Operable status within 7 days.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, 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 April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination in its application dated October 24, 2003.

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

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

A change is proposed to allow the affected SDV vent and drain line to be isolated when there are one or more SDV vent or drain lines with one valve inoperable instead o[f] requiring the valve to be restored to operable status within 7 days. With one SDV vent or drain valve inoperable in one or more lines, the isolation function would be maintained since the redundant valve in the affected line would perform its safety function of isolating the SDV. Following the completion of the required action, the isolation function is fulfilled since the associated line is isolated. The ability to vent and drain the SDVs is maintained and controlled through administrative controls. This requirement assures the reactor protection system is not adversely affected by the inoperable valves. With the safety functions of the valves being maintained, the probability or consequences of an accident previously evaluated are not significantly increased.

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

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

The proposed change ensures that the safety functions of the SDV vent and drain valves are fulfilled. The isolation function is maintained by redundant valves and by the required action to isolate the affected line. The ability to vent and drain the SDVs is maintained through administrative controls. In addition, the reactor protection system will prevent filling of an SDV to the point that it has insufficient volume to accept a full scram. Maintaining the safety functions related to isolation of the SDV and insertion of control rods ensures that the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 31, 2003, as supplemented on October 10, 2003.

Description of amendment request: This amendment request incorporates a revision to the licensing basis of the Vermont Yankee Nuclear Power Station (VYNPS) that supports a full scope application on an Alternative Source Term (AST) methodology.

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

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

Adoption of the AST and those plant systems affected by implementation of the AST do not initiate DBAs [design basis accidents]. The proposed change does not affect the design or manner in which the facility is operated; rather, once the occurrence of an accident has been postulated, the new accident source term is an input to analyses that evaluate the radiological consequences. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The structures, systems and components (SSCs) affected by the proposed change act as mitigators to the consequences of accidents. Based on the revised analyses, the proposed changes do revise certain performance requirements; however, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a design basis accident discussed in Chapter 14 of the Updated Final Safety Analysis Report.

Because of the changed methodology, it is difficult to draw a quantitative comparison of before and after accident consequences due to the use of different dose calculations, conversion factors, source term, and other assumptions. However qualitatively, it can be shown that there is no significant increase in offsite doses, although there may be small variations in potential doses for postulated accidents. Plant-specific radiological analyses have been performed using the AST methodology. Based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the offsite doses are well within acceptable limits. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan (SRP) Section 15.0.1.

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

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

Implementation of AST and the proposed changes does not alter or involve any design basis accident initiators. These changes do not affect the design function or mode of operations of SSCs in the facility prior to a postulated accident. Since SSCs are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

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

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

The changes proposed are associated with a revision to the licensing basis for the VYNPS. Approval of the licensing basis change from the original source term to the alternative source term is requested by this application for a license amendment. The results of the accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183, and SRP 15.0.1. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 17, 2003.

Description of amendment request: The proposed amendments revise the action requirements of Technical Specification (TS) 3/4 6.3, ''Containment Isolation Valves [CIVs],'' to more clearly define action requirements for inoperable CIVs. The proposed changes to the action requirements also include: (1) Provisions for allowing the intermittent unisolation of penetration flow paths which have been isolated per action requirements under administrative control; (2) use of check valves as an isolation device; and (3) an increase in the allowed outage time to 72 hours for CIVs associated with closed systems inside containment. The proposed amendments also revise the TS surveillance requirements (SRs) for CIVs by replacing existing SRs with new SRs similar to those in NUREG–1431, Revision 2, ''Standard Technical Specifications for Westinghouse Plants.''

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

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

Response: No.

The proposed change does not involve any changes to plant equipment, system design functions or a change in the methods governing normal plant operation. Therefore, the probability of a malfunction of a structure, system or component to perform its design function will not be increased.

The proposed change modifies existing action requirements for inoperable containment isolation valves. Action requirements and their associated allowed outage times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that repaired equipment is out of service prior to the analyzed event. In addition, changes that are consistent with the ISTS [improved Standard Technical Specifications] have been previously evaluated and found not to adversely affect the safe operation of Westinghouse plants or the initiation of any accident previously evaluated. Based on the conclusions of the plant specific evaluation associated with the changes and the evaluation performed in developing the ISTS, the proposed revised action requirements do not result in operating conditions that will significantly increase the probability of initiating an analyzed event. The revised action requirements provide appropriate remedial actions to be taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased.

The proposed change also modifies and deletes some surveillance requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment specified in the Limiting Condition for Operation is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. This equipment will continue to be tested in a manner and at a frequency to give confidence that the equipment can perform its assumed safety function. The proposed changes are generally made to conform to the ISTS and have been evaluated to not be detrimental to plant safety. As a result, the proposed surveillance requirement changes do not significantly affect the consequences of any accident previously evaluated. 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 change does not involve any changes to plant equipment, system design functions or a change in the methods governing normal plant operation. The [technical] specification for containment isolation valves provide[s] controls for maintaining the containment pressure boundary. The revised action requirements and revised surveillance requirements are sufficient to ensure the containment isolation valves are capable of performing their accident mitigation functions. No new accident initiators are introduced by these changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The revised action requirements do not involve a significant reduction in the margin of safety. The proposed actions for inoperable containment isolation valves minimize the risk of continued operation under the specified conditions, considering the operability status of the redundant containment isolation barriers, a reasonable time for repairs or replacement of the isolation feature, and the low probability of a design basis accident occurring during the repair period.

The revised surveillance requirements do not involve a significant reduction in the margin of safety. The proposed surveillance requirements provide the required verifications for ensuring containment isolation valves operability. Containment isolation valve testing will continue to be performed in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function.

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

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

Attorney for licensee: Mary 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: December 17, 2001, as supplemented by letter dated June 4, 2002.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.3.1, ''Reactor Coolant System Instrumentation,'' to delete an action involving either reducing core thermal power and the high neutron flux reactor trip setpoint or monitoring quadrant power tilt when a reactor protection system (RPS) channel is inoperable. Additionally, changes to the content and format of TS Tables 3.3–1 and 4.3–1 are proposed to enhance specification clarity.

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

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

The proposed change does not result in an increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. The proposed change does not result in an increase in the consequences of an accident previously evaluated because TS 3/4.2.4, ''Quadrant Power Tilt,'' continues to ensure the radial power distribution of the core is within the limits assumed in the accident analyses. In addition, compensatory actions will continue to be required should a single channel of RPS High Flux or Flux-'Flux-Flow become inoperable. 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 changes affect the TS requirements for the RPS instrumentation. The proposed changes do not change the RPS design function or result in the RPS being operated outside its design operating range. There are no new or different equipment failure modes introduced by the proposed changes. 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. The proposed changes do not involve a significant reduction in a margin of safety.

The proposed changes affect the TS requirements for the RPS instrumentation. The capability of the RPS to perform its required functions is not adversely affected by the proposed changes. The proposed changes do not alter any initial conditions contributing to accident severity or consequences. There will be no changes to the plants' systems, structures, or components, nor in the manner in which they will be operated as a result of the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine.

Date of amendment request: September 11, 2003.

Description of amendment request: Revise the dose model for the containment activated concrete, rebar (hereafter referred to as activated concrete) and liner, by incorporating more realistic radionuclide release rates and to change the associated derived concentration guideline limit (DCGL) for activated concrete.

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

Response: No.

The requested license amendment does not authorize any plant activities beyond those allowed by 10 CFR Chapter I or beyond those considered in the DSAR. The bounding accident described in the Defueled Safety Analysis Report (DSAR) for potential airborne activity is the postulated resin cask drop accident in the Low Level Radioactive Waste Storage Building. This accident is expected to contain more potential airborne activity than can be released from other decommissioning events. The radionuclide distribution assumed for the spent resin cask has a greater inventory of transuranic radionuclides (the major dose contributor) than the distribution of plant derived radionuclides in the components involved in other decommissioning accidents. The other accidents considered in the DSAR include: (1) Explosion of liquid petroleum gas (LPG) leaked from a front end loader or forklift; (2) Explosion of oxyacetylene during segmenting of the reactor vessel shell; (3) Release of radioactivity from the RCS decontamination ion exchange resins; (4) Gross leak during in-situ decontamination; (5) Segmentation of RCS piping with unremoved contamination; (6) Fire involving contaminated clothing or combustible waste; (7) Loss of local airborne contamination control during blasting or jackhammer operations; (8) Temporary Loss of Services; (9) Dropping of Contaminated Concrete Rubble; (10) Natural phenomena; and (11) Transportation accidents. The probabilities and consequences for these accidents are estimated in the basis documentation for DSAR Section 7. No systems, structures, or components that could initiate or be required to mitigate the consequences of an accident are affected by the proposed change in any way not previously evaluated in the DSAR. Since Maine Yankee does not exceed the salient parameters associated with the plant referenced in the basis documentation in any material respects, it is concluded that these probabilities and consequences are not increased. Therefore, the proposed change to the Maine Yankee license does not involve any increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

The requested license amendment does not authorize any plant activities that could precipitate or result in any accidents beyond those considered in the DSAR. The accidents previously evaluated in the DSAR are described above. These accidents are described in the basis documentation for DSAR Section 7. The proposed change does not affect plant systems, structures, or components in any way not previously evaluated in the DSAR. Since Maine Yankee does not exceed the salient parameters associated with the plant referenced in the basis documentation in any material respects, it is concluded that these accidents appropriately bound the kinds of accidents possible during decommissioning. Therefore, the proposed change to the Maine Yankee license would 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?

Response: No.

The margin of safety defined in Maine Yankee's license basis for the consequences of decommissioning accidents has been established as the margin between the bounding decommissioning accident and the dose limits associated with the need for emergency plan offsite protection, namely the Environmental Protection Agency Protective Action Guidelines EPA-PAGs. As described above, the bounding decommissioning accident is the postulated resin cask drop accident in the Low Level Radioactive Waste Storage Building. Since the bounding decommissioning accident is expected to contain more potential airborne activity than can be released from other decommissioning events and since the radionuclide distribution assumed for the spent resin cask has more transuranics (the major dose contributor) than the distribution in the components involved in other decommissioning accidents, the margin of safety associated with the consequences of decommissioning accidents cannot be reduced. The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use. This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Since the license termination plan (LTP) was designed to comply with the radiological criteria for license termination for unrestricted use, the margin of safety cannot be reduced. Therefore, the proposed changes to the Maine Yankee license would not involve a significant reduction in any margin of safety.

Conclusion

Based on the above, Maine Yankee 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 amendment involves no significant hazards consideration.

Attorney for licensee: Joe Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: September 26, 2003.

Description of amendment request: The proposed amendments would modify TS 5.6.5.b to add a reference to a Nuclear Regulatory Commission (NRC) letter that would approve the use of a new master curve methodology for Unit 2. The NRC staff is currently reviewing an associated exemption request by NMC to use this new methodology. The requested exemption would allow the use of the master curve methodology described in Babcock & Wilcox Report BAW–2308, Revision 1, ''Initial RT_{NDT} [reference nil-ductility temperature] of Linde 80 Weld Materials,'' for determining the adjusted RT_{NDT} of the Unit 2 reactor vessel limiting circumferential weld metal. This method is used for the pressurized thermal shock screening evaluation. The proposed amendments would also make editorial changes to TS 5.6.5.b.

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

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

The proposed change references the NRC safety evaluation [currently under NRC staff review] accepting the new Master Curve Methodology used in the evaluation of the revised P/T [pressure/temperature] limits and LTOP [low-temperature overpressure protection] setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences. Therefore, it is concluded that this change does not increase the probability of occurrence of an accident previously evaluated.

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

The proposed change references the NRC safety evaluation [currently under NRC staff review] accepting the new Master Curve Methodology used in the evaluation of the revised P/T limits and LTOP setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The change does not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of PBNP in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The proposed change references the NRC safety evaluation [currently under NRC staff review] accepting the new Master Curve Methodology used in the evaluation of the revised P/T limits and LTOP setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

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: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

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: September 12, 2003.

Description of amendment requests: The proposed license amendments would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' and TS 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' to change the current steam generator (SG) narrow range (NR) water level-low low setpoints from greater than or equal to 7.0 percent allowable value and 7.2 percent nominal value, to greater than or equal to 14.8 percent allowable value and 15.0 percent nominal value.

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

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

The protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes and the actuation logic changes are conservative. The design of the steam generator (SG) water level sensing equipment and the coincidence logic will be unaffected. The only physical change to the reactor trip system (RTS) and the engineered safety feature actuation system (ESFAS) instrumentation is the increased actuation setpoints. These changes have already been implemented in the plant through the design change process. These changes are in the conservative direction, *i.e.*, a trip actuation signal will be generated sooner for an event that challenges the ability of the SGs to provide a heat sink for the reactor. In all other regards, the design of the RTS and ESFAS instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis.

The probability and consequences of accidents previously evaluated in the Final Safety Analysis Report Update (FSARU) are not adversely affected because changes to the RPS and ESFAS trip setpoints assure a conservative response of the affected trip functions, consistent with the safety analyses and licensing basis.

The proposed changes will not affect the probability of any accident initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSARU.

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

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

The proposed changes do not change any hardware or the design functions of any structures, systems or components involved, other than to revise the SG narrow range (NR) water level-low low setpoints; changes that have already been implemented. The proposed changes will not affect the normal method of plant operation or change any operating parameters. No new accidents, accident initiators, or failure mechanisms are created by the proposed changes.

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 SG NR water level-low low setpoints specified in the Technical Specifications have already been increased in the conservative direction. The safety analysis limits assumed in the transient and accident analyses remain unchanged. None of the acceptance criteria for any accident analysis are changed.

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.

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: October 22, 2003.

Description of amendment requests: The proposed license amendments would revise Surveillance Requirement 3.6.3.7 of Technical Specification (TS) 3.6.3, ''Containment Isolation Valves,'' by extending the leakage rate testing frequency of the containment purge supply and exhaust and vacuum/pressure relief valves, all with resilient seals, from 184 days to 24 months. The amendments would also delete the requirement to leakage rate test the containment vacuum/pressure relief valves within 92 days after opening.

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.

Operability and leakage control effectiveness of the containment purge supply and exhaust and containment vacuum/pressure relief isolation valves have no effect on whether an accident occurs. Consequently, increasing the interval between surveillances of isolation valve leak rate does not involve any significant increase in the probability of an accident previously evaluated. The consequences of a unisolated reactor containment building at the time of a fuel-handling accident or loss of coolant accident (LOCA) are the release of radionuclides to the environment. Offsite exposures due to containment leakage during a LOCA and fuel-handling accident have been evaluated in Final Safety Analysis Report Update (FSARU) sections 15.5.17.3 and 15.5.22, respectively. For a LOCA, the Diablo Canyon Power Plant (DCPP) analyses assume containment leakage of 0.1 percent of the containment volume per day for the first 24 hours and 0.05 percent per day for the rest of the duration of the accident. Calculated radiological exposures from the LOCA are listed in FSARU Chapter 15, Table 15.5–75 and are within the 10 CFR part 100 limits. The good performance history of these valves, along with the very low total containment leakage rate, are reasonable bases that there should not be any significant increase in the consequences of [an] accident previously evaluated. For the fuel-handling accident inside containment, DCPP analyses do not credit these valves to provide a containment isolation function. It was assumed that activity released from the containment refueling pool is transported to the environment over a short time period through the open equipment hatch. Calculated radiological exposures from the fuel-handling accident inside containment are listed in FSARU Chapter 15, Table 15.5–50 and are also within the 10 CFR part 100 limits. In summary, increasing the interval between leakage rate surveillances of these isolation valves will not involve any significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a modification to the physical configuration of the plant (*i.e.*, no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The functions of the containment purge and containment vacuum/pressure relief systems are not altered by this change. Therefore, the proposed change does 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.

This proposed change only increases the interval between surveillance tests of the containment purge supply and exhaust, and containment vacuum/pressure relief valves. These valves have a good performance history and should be able to perform their intended containment isolation function reliably when called upon. In FSARU Chapter 15, two offsite exposure scenarios are applicable to the containment isolation function. These scenarios are LOCA containment leakage and fuel-handling accident inside containment. For LOCA containment leakage, the DCPP analyses assume containment leakage of 0.1 percent of the containment volume per day for the first 24 hours and 0.05 percent per day for the remainder of the accident. Calculated radiological exposures from a LOCA are listed in FSARU Chapter 15, Table 15.5–75 and meet the 10 CFR part 100 limits. For the fuel-handling accident inside containment, the DCPP analyses do not credit these valves to provide a containment isolation function. The analyses assume that activity released from the containment refueling pool is transported to the environment over a short time period through the open equipment hatch. Calculated radiological exposures from the fuel-handling accident inside containment are listed in FSARU Chapter 15, Table 15.5–50 and also meet the 10 CFR part 100 limits. If in the unlikely event that these valves exceed their leakage rate limits due to the extension of the surveillance interval, the consequences will be consistent with the containment leakage assumed in the accident analyses. Therefore, the extension of leakage rate test interval will have an insignificant radiological consequence, and the proposed change will not involve any significant reduction in the margin of safety.

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

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: October 22, 2003.

Description of amendment requests: The proposed license amendments would revise Technical Specifications (TS) Section 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and TS Section 5.6.10, ''Steam Generator (SG) Tube Inspection Report,'' to allow use of leak limiting Alloy 800 sleeves to repair degraded SG tubes as an alternative to plugging the SG tubes. The proposed amendments would also remove an unnecessary reporting requirement contained in TS Table 5.5.9–2, ''Steam Generator (SG) Tube Inspection.''

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 leak limiting Alloy 800 sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator (SG) tubing. The applied stresses and fatigue usage for the sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes.'' Burst testing of sleeve-tube assemblies has confirmed the analytical results and demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The leak limiting Alloy 800 sleeve depth-based structural limit is determined using NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for the configuration of long axial cracks. A sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly.

Evaluation of the repaired SG tube testing and analysis indicates no detrimental effects on the leak limiting Alloy 800 sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Diablo Canyon Power Plant (DCPP) Units 1 and 2. Corrosion testing and historical performance of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed change has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the leak limi[ti]ng Alloy 800 sleeve-tube assembly is bounded by the current SG tube rupture (SGTR) analysis described in the DCPP Final Safety Analysis Report Update. Due to the slight reduction in the inside diameter caused by the sleeve wall thickness, primary coolant release rates through the parent tube would be slightly less than assumed for the SGTR analysis and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feedwater line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the DCPP safety analysis. The sleeve-tube assembly leakage during plant operation would be minimal and is well within the Technical Specification (TS) leakage limits.

The proposed change to TS 5.5.9 Table 5.5.9–2, ''Steam Generator (SG) Tube Inspection,'' to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) if the first sample inspection or the second sample inspection results in a C–3 classification, is an administrative change only and does not affect plant equipment or accident analyses.

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 leak limiting Alloy 800 sleeves are designed using the applicable ASME Code as guidance, and therefore meet the objectives of the original SG tubing. As a result, the functions of the SG will not be significantly affected by the installation of the proposed sleeve. The proposed sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR accident analysis. The continued integrity of the installed sleeve-tube assembly is periodically verified by the TS requirements and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly.

Implementation of the proposed change has no significant effect on either the configuration of the plant, or the manner in which it is operated. The proposed change to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) from TS 5.5.9 Table 5.5.9–2 is an administrative change only and does not affect plant equipment or accident analyses.

Therefore, the proposed change does 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 repair of degraded SG tubes with leak limiting Alloy 800 sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions and thereby maintains current core cooling margin as opposed to plugging the tube and taking it out of service. The design safety factors utilized for the sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original SG design. The sleeve and portions of the installed sleeve-tube assembly that represent the reactor coolant pressure boundary will be monitored and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not significantly different from the original SG tubes.

The proposed change to delete the requirement to notify the NRC pursuant to 10 CFR 50.72(b)(2) from TS 5.5.9 Table 5.5.9–2 is an administrative change only, does not affect plant equipment or accident analyses, does not relax any safety system settings, and does not relax the bases for any limiting conditions for operations.

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.

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: November 4, 2003.

Description of amendment request: The proposed amendments would revise the South Texas Project, Units 1 and 2 Technical Specifications for the Remote Shutdown System to reflect requirements consistent with those in NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants.''

The proposed changes would increase the allowed outage time for inoperable Remote Shutdown System components to a time that is more consistent with their safety significance. It would also relocate the description of the required components to the Bases where it will be directly controlled by the licensee.

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.

Because the proposed changes do not involve potential accident initiators, there is no significant increase in the probability of an accident previously evaluated. There is no proposed change to the design basis or configuration of the plant and the extension of the allowed outage time of the Remote Shutdown System functions does not have a significant effect on safety. Consequently there is no significant increase in the 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 affect how the plant is operated or involve any physical changes to the plant. Therefore there is no 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.

Except for extending the allowed outage time for Remote Shutdown System function from 7 days to 30 days, the proposed changes are essentially administrative. The evaluation of the extension of the allowed outage time demonstrated that there was no 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 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.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: June 24, 2003.

Brief description of amendment: The amendment revises Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to Operable status within 7 days.

Date of issuance: October 30, 2003.

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

Amendment No.: 157.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 19, 2003 (68 FR 49815).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2003.

No significant hazards consideration comments received: No.

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

Date of application of amendments: July 10, 2003.

Brief description of amendments: The amendments revised the Technical Specifications to remove requirements that are no longer applicable because the implementation of the automatic feedwater isolation system modification has been completed on all three Oconee units.

Date of Issuance: November 5, 2003.

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

Amendment Nos.: 336, 336, & 337.

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

Date of initial notice in ***Federal Register:*** August 19, 2003 (68 FR 49816). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 5, 2003.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi.

Date of application for amendment: April 3, 2003.

Brief description of amendment: The changes revise the Updated Final Safety Analysis Report to change the Reactor Vessel Material Surveillance Program. The change reflects participation in the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program.

Date of issuance: November 4, 2003.

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

Amendment No: 160.

Facility Operating License No. NPF–29: The amendment revises the Updated Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** May 13, 2003 (68 FR 25653).

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

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York.

Date of application for amendment: October 23, 2001, as supplemented on March 29 and December 17, 2002, and June 12, 2003.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.5.10, ''Ventilation Filter Testing Program,'' to adopt the requirements of the American Society for Testing and Materials Standard D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' The TS revisions are in response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The amendment revises the TSs: (1) To provide a control room ventilation system (CRVS) methyl iodide removal efficiency of greater than or equal to 95.5% and remove the notation that there is a 1-inch charcoal bed depth; (2) to allow for the continued use of the existing CRVS through Refueling Outage 13, in order to design, fabricate, and install a 2-inch charcoal filter bed; and (3) to add a note in the TS requiring a demonstration of charcoal efficiency of 93% when changing the charcoal in the existing CRVS bed prior to any fuel movement in the upcoming Refueling Outage 12 and every 6 months thereafter until the new beds are installed. The NRC had previously published a notice of consideration on December 12, 2001 (66 FR 64292) regarding a similar proposal from the licensee in response to GL 99–02. However, in response to a request for additional information from the NRC dated March 29, 2002, the licensee revised its application and withdrew the prior request to change the maximum CRVS differential pressure in TS 5.5.10.d.

Date of issuance: October 30, 2003.

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

Amendment No.: 219.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 18, 2003 (68 FR 12951).

The March 29 and December 17, 2002, and June 12, 2003, letters provided clarifying information that did not enlarge the scope of the amendment request or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 16, 2002, as supplemented by letters dated July 30, and September 29, 2003.

Brief description of amendment: The amendment adds Combustion Engineering topical report CEN–372–P–A, May 1990, ''Fuel Rod Maximum Allowable Gas Pressure,'' to the list of topical reports in Technical Specification 6.9.1.11.1, used to determine the Waterford Steam Electric Sation, Unit 3 core operating limits. In addition, the amendment approves the deletion of applicable dates and revision numbers for CEN–372–P–A and other topical reports listed in TS 6.9.1.11.1.

Date of issuance: October 31, 2003.

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

Amendment No.: 191.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 4, 2003 (68 FR 5673). The July 30, and September 29, 2003, 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 amendment is contained in a Safety Evaluation dated October 31, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 31, 2003.

Brief description of amendments: The amendments revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the changes modify TS 5.7, ''High Radiation Area,'' by incorporating the wording and requirements from NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001. The revision also includes administrative changes regarding access control and terminology for high radiation areas.

Date of issuance: October 31, 2003.

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

Amendment Nos.: 161/147.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 27, 2003 (68 FR 28852).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353.

Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania.

Date of application for amendments: December 20, 2002, as supplemented May 30, 2003.

Brief description of amendments: The amendments removed the current facility reactor material specimen surveillance schedule from the Technical Specifications for Limerick Generating Station, Units 1 and 2 (LGS–1 and 2). The licensee also revised the Updated Final Safety Analysis Report (UFSAR) for LGS–1 and 2 to reflect implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program as the basis for demonstrating the compliance with the requirements of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' to title 10 of the Code of Federal Regulations, Part 50.

Date of issuance: November 4, 2003.

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

Amendment Nos.: 167 and 130.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications and authorized changes to the UFSAR for LGS–1 and 2.

Date of initial notice in **Federal Register:** February 4, 2003 (68 FR 5669). The supplement dated May 30, 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 amendments is contained in a Safety Evaluationdated November 4, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC,

Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station,

Units 2 and 3, (PBAPS-2 and 3) York County and Lancaster County, Pennsylvania.

Date of application for amendments: December 20, 2002, as supplemented May 30, 2003.

Brief description of amendments: The amendments revised the Updated Final Safety Analysis Report (UFSAR) for Peach Bottom Atomic Power Station, Units 2 and 3, by allowing implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program as the basis for demonstrating the compliance with the requirements of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' to Title 10 of the Code of Federal Regulations, Part 50.

Date of issuance: November 4, 2003.

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

Amendment Nos.: 249 and 253.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments authorized changes to the UFSAR for PBAPS–2 and 3.

Date of initial notice in **Federal Register:** February 4, 2003 (68 FR 5669). The supplement dated May 30, 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 amendments is contained in a Safety Evaluation dated November 4, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 27, 2003, as supplemented August 15, 2003.

Brief description of amendment: The amendment lowers the trip setpoint and allowable value contained in Technical Specification (TS) Table 3.3–4 for the pressurizer pressure low safety injection signal. The amendment also lowers the value for the P–11 setpoint in TS Table 3.3–3. These changes increase the margin between the low pressurizer pressure safety injection actuation setpoint and the minimum pressurizer pressure that occurs immediately following a reactor trip.

Date of issuance: November 12, 2003.

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

Amendment No.: 263.

Facility Operating License No. DPR–74: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** May 27, 2003 (68 FR 28853).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 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: December 31, 2002, as supplemented by letter dated July 24, 2003.

Brief description of amendment: The amendment revises the Updated Safety Analysis Report (USAR) reflecting a change of the reactor vessel material surveillance program to incorporate the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program into the licensing basis.

Date of issuance: October 31, 2003.

Effective date: As of the date of issuance. The amendment shall be implemented within 30 days of issuance and the USAR changes shall be implemented in the next periodic update to the USAR in accordance with 10 CFR 50.71(e).

Amendment No.: 201.

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

Date of initial notice in **Federal Register:** February 4, 2003 (68 FR 5678).

The July 24, 2003, supplemental letter 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 as published in the **Federal Register** on February 4, 2003 (68 FR 5678).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: January 27, 2003, as supplemented by letter dated August 1, 2003.

Brief description of amendment: The amendment authorizes revisions to the Updated Safety Analysis Report (USAR) to incorporate the NRC approval of the GOTHIC 7.0 computer program for performing containment analyses.

Date of issuance: November 5, 2003.

Effective date: November 5, 2003, and shall be implemented within 30 days of the date of issuance. The implementation of the amendment includes the incorporation into the USAR the changes discussed above, as described in the licensee's application dated January 27, 2003, and supplement dated August 1, 2003, and evaluated in the staff's Safety Evaluation attached to the amendment.

Amendment No.: 222.

Renewed Facility Operating License No. DPR–40: The amendment revised the USAR.

Date of initial notice in **Federal Register:** March 18, 2003 (68 FR 12956).

The August 1, 2003, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: January 27, 2003, as supplemented by letter dated October 14, 2003.

Brief description of amendment: The amendment deletes Technical Specification (TS) 2.3(2)i and the corresponding Bases that allows the performance of the surveillance test of Table 3–2, Item 20 (Recirculation Actuation Logic Channel Functional Test) under administrative controls, while components in excess of those allowed by Conditions a, b, d, and e of TS 2.3(2) are inoperable, provided they are returned to operable status within one hour. This allowance was granted in Amendment No. 206 issued April 19, 2002, and only applied until the end of Cycle 21.

Date of issuance: November 10, 2003.

Effective date: November 10, 2003, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 223.

Renewed Facility Operating License No. DPR–40: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 18, 2003 (68 FR 12955).

The October 14, 2003, supplemental letter 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 November 10, 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: May 6, 2003, as supplemented by letters dated August 12 and September 18, 2003.

Brief description of amendments: These amendments deleted Technical Specification (TS) 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' and revised TS 3.4.1, ''Recirculation Loops Operating,'' to formally extend the currently implemented requirements, which define appropriately conservative restrictions to plant operation and operator response to thermal hydraulic instability events. In addition, the amendments revise TS 3.4.1 to refer to the power flow map in the core operating limits report and include a reference in TS 5.6.5.

Date of issuance: October 29, 2003.

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

Amendment Nos.: 215 and 190.

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

*Date of initial notice in **Federal Register***: June 24, 2003 (68 FR 37582).

The supplemental letters dated August 12 and September 18, 2003, provided clarifying information that did not change the scope of the amendment as described in the initial notice of the proposed action published in the **Federal Register** notice (68 FR 37582, June 24, 2003), or the U.S. Nuclear Regulatory Commission staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 25, 2003.

Description of amendment request: The amendments revised Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to operable status within 7 days.

Date of issuance: November 3, 2003.

Effective date: Date of issuance, to be implemented within 60 days.

Amendment Nos.: 248, 285, and 243.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68. Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 18, 2003 (68 FR 54753).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 17th day of November, 2003.

For the Nuclear Regulatory Commission.

Eric Leeds,

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

[FR Doc. 03–29107 Filed 11–24–03; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Federal Employees' Group Life Insurance Program: New Option B Premiums

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: The Office of Personnel Management (OPM) is announcing new Federal Employees' Group Life Insurance (FEGLI) premiums for the upper age bands of Option B. The premiums will be maintained on the FEGLI Web site at *http://www.opm.gov/insure/life.*

EFFECTIVE DATE: January 1, 2004.

FOR FURTHER INFORMATION CONTACT: Karen Leibach, (202) 606–0004.

SUPPLEMENTARY INFORMATION: On December 30, 2002, OPM published a **Federal Register** notice (67 FR 79659) announcing premium changes for FEGLI and new age bands for Options B and C. The premiums for the new Option B age bands are being phased in over a 3-year period. The first set of premiums for these age bands was effective the first pay period beginning on or after January 1, 2003.

This notice announces the second phase of the Option B premium changes. These premiums are effective the first pay period beginning on or after January 1, 2004.

OPTION B PREMIUM PER $1,000 OF INSURANCE

Age band	Biweekly	Monthly
70–74	$1.03	$2.232
75–79	1.43	3.098
80 and over	1.83	3.965

The premiums for compensationers, who are paid every 4 weeks, are 2 times the biweekly premium amounts.

Premiums for other FEGLI coverages, including premiums for other Option B age bands, are not changing.

U.S. Office of Personnel Management.

Kay Coles James,

Director.

[FR Doc. 03–29438 Filed 11–24–03; 8:45 am]

BILLING CODE 6325–50–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–48800; File No. SR–Amex–2002–116]

Self-Regulatory Organizations; Notice of Filing of a Proposed Rule Change and Amendment Nos. 1, 2, 3, and 4 Thereto by the American Stock Exchange LLC Relating to Specialist Stabilization Requirements for Derivative Products

November 17, 2003.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] notice is hereby given that on December 27, 2002, the American Stock Exchange LLC (''Amex'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in

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

[2] 17 CFR 240.19b–4.