costs, and assessing the Medicare benefit package.

Agendas will be mailed on January 8, 2002. The final agenda will be available on the Commission's Web site (*www.MedPAC.gov*).

ADDRESSES: MedPAC's address is: 1730 K Street, NW, Suite 800, Washington, DC 20006. The telephone number is (202) 653–7220.

FOR FURTHER INFORMATION CONTACT: Diane Ellison, Office Manager, (202) 653–7220.

Murray N. Ross,

Executive Director.

[FR Doc. 02–368 Filed 1–7–02; 8:45 am]

BILLING CODE 6820–BW–M

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice: (02–002)]

Notice of Agency Report Forms Under OMB Review

AGENCY: National Aeronautics and Space Administration (NASA).

ACTION: Notice of Agency report forms under OMB review.

SUMMARY: The National Aeronautics and Space Administration, as part of its continuing effort to reduce paperwork and respondent burden, invites the general public and other Federal agencies to take this opportunity to comment on proposed and/or continuing information collections, as required by the Paperwork Reduction Act of 1995 (Public Law 104–13, 44 U.S.C. 3506(c)(2)(A)). This information collection provides records of accountability, responsibility, transfer, location, and disposition of radioactive materials.

DATES: Comments on this proposal should be received within 30 calendar days from the date of this publication.

ADDRESSES: All comments should be addressed to Desk Officer for NASA; Office of Information and Regulatory Affairs; Office of Management and Budget; Room 10236; New Executive Office Building; Washington, DC, 20503.

FOR FURTHER INFORMATION CONTACT: Ms. Nancy Kaplan, NASA Reports Officer, (202) 358–1372.

Title: Radioactive Material Transfer Receipt.

OMB Number: 2700–0007.

Type of review: Extension.

Need and Uses: NASA Johnson Space Center is required by federal law to keep records of the receipt, transfer, and disposal of radioactive items and information on accountability, responsibility, transfer, disposition, and location.

Affected Public: Business or other for-profit; Federal Government, state, local or tribal government.

Number of Respondents: 25.

Responses Per Respondent: 2.

Annual Responses: 50.

Hours Per Request: Approximately ½ hr.

Annual Burden Hours: 29.

Frequency of Report: On occasion.

David B. Nelson,

Deputy Chief Information Officer, Office of the Administrator.

[FR Doc. 02–378 Filed 1–7–02; 8:45 am]

BILLING CODE 7510–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 7, 2001, through December 27, 2001. The last biweekly notice was published on December 26, 2001 (66 FR 64461).

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 may be examined at the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By February 7, 2002 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor) Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Document 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 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 Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor) Rockville, Maryland, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and 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 Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor) Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm./adams.html*. Persons who 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 *pdc@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: November 19, 2001.

Description of amendments request: The proposed amendments would change the Loss of Feedwater Flow analysis in the Updated Final Safety Analysis Report (UFSAR). The current analysis contained several non-conservative assumptions, which would be corrected by the reanalysis. These corrections include: incorporating a single-failure of the Auxiliary Feedwater System, including steam generator blowdown, accounting for the change in density of water once the feedwater flow has stopped, and assuming sludge deposition in the steam generators. Also assumed in the analysis is the installation of a modification to isolate steam generator blowdown on an auxiliary feedwater actuation signal and an operator action to adjust the auxiliary feedwater flow after the event initiation.

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

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

No. The proposed amendments would modify several assumptions in the UFSAR Loss of Feedwater Flow analysis to more accurately reflect the plant response to the event. The significant changes to the accident include the addition of an operator action to increase auxiliary feedwater (AFW) flow and the implementation of an automatic steam generator blowdown isolation following the event initiation. These changes do not affect any accident initiators or precursors because they only alter the operation of the plant following the accident initiation. Thus, the proposed amendments do not increase the probability of occurrence of any previously analyzed accidents. In addition, besides the aforementioned changes, the proposed amendments would not alter any design parameter, condition, equipment configuration, or manner in which Calvert Cliffs Units 1 and 2 are operated. Furthermore, the proposed modifications do not alter or prevent the ability of existing structures, systems, or components to perform their intended safety or accident-mitigating functions depicted in the UFSAR. The proposed modifications to the analysis account for a single-failure and update the assumptions to more recent standards. With these changes, the plant continues to meet the current acceptance criteria. Therefore, the proposed amendments do not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed amendments alter the design function of the steam generator blowdown valves to isolate upon receipt of an auxiliary feedwater actuation system signal. The isolation of the steam generator blowdown will not create the possibility of a new or a different kind of accident because blowdown isolation already occurs on a high radiation signal or a containment spray actuation signal. The operator action to increase AFW flow only alters the operation of the AFW in the conservative direction. The other changes to the accident analysis do not alter any design parameter, condition, equipment configuration, or manner in which the units are operated. Furthermore, none of the changes alter or prevent the ability of structures, systems, or components to perform their intended safety or accident mitigating functions. Accordingly, the proposed amendment does not create any new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendments do not change any design parameter, safety limit, or acceptance criteria. However, the amendments do change an analysis methodology. This change, performed with the objective of imposing conservative assumptions on the accident analysis, keeps the accident within the acceptance criteria for anticipated operational occurrences. Therefore, operation in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan, Acting.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of amendment request: November 19, 2001.

Description of amendment request: The proposed amendment would provide a one-time extension, from 10 to 14 days, of the allowed outage time (AOT) for one train of the Control Room Emergency Ventilation System (CREVS) to be inoperable due to the emergency power supply being inoperable.

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

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

No. The proposed amendment would allow a one-time extension of the AOT of one train of the CREVS. Since the CREVS is an accident mitigation system, the AOT extension would not affect any accident initiators or precursors. Therefore, the AOT extension would not increase the probability of an accident previously evaluated. Similarly, since the consequences of a design-basis accident coincident with a failure of the redundant CREVS train during a 14-day outage are the same as those during the already approved 10-day outage, the proposed change does not increase the consequences of an accident previously evaluated.

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

No. The proposed amendment does not affect accident initiators or precursors because the CREVS is not being modified nor will any unusual operator actions be required. The CREVS is not an accident initiator, but is an accident mitigator. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment would allow a one-time extension of the AOT of one train of the CREVS from the currently allowed 10 days to 14 days. This action decreases the margin of safety. However, based upon the licensee's management of plant risk, the change increases the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) to less than the Regulatory Guide 1.174 criteria of 1E–6 per reactor year for CDF and 1E–07 per reactor year for LERF. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendments request: November 26, 2001.

Description of amendments request: The proposed amendments will add Technical Specification 5.5.12.f, ''Programs and Manuals, Primary Containment Leakage Rate Testing Program.'' This addition will provide a one-time exception to the frequency of the performance-based leakage rate testing program.

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

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

The proposed change to Technical Specification 5.5.12 provides a one-time extension to the testing frequency for containment integrated leakage rate (i.e., Type A) testing. The existing 10-year test interval is based on past test performance. The proposed Technical Specification change will extend the Type A testing frequency to 15 years, one month from the last Type A test for Unit 1 and to 15 years for Unit 2. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the primary containment does not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only a one-time change to the interval between Type A containment leakage tests. Type B and C containment leakage testing will continue to be performed at the frequency currently required by the BSEP Technical Specifications. As documented in NUREG–1493, ''Performance-Based Containment Leakage-Test Program,'' industry experience has shown that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. In fact, an analysis of 144 integrated leak rate tests results, including 23 failures, found that no failures were due to containment liner breach. NUREG–1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate testing to once per 20 years was found to lead to an imperceptible increase in risk. The BSEP, Unit 1 and 2 test history and risk-based evaluation of the proposed extension to the Type A test frequency supports this conclusion. The design and construction requirements of the primary containment, combined with the containment inspections performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI and the Maintenance Rule (i.e., 10 CFR 50.65) provide a high degree of assurance that the primary containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change to the Technical Specification 5.5.12 involves a one-time extension to the testing interval for Type A containment leakage rate testing. The primary containment and the testing requirements invoked to periodically demonstrate the integrity of the primary containment exist to ensure the ability to mitigate the consequences of an accident. The primary containment and its associated testing requirements do not involve the prevention or identification of any precursors of an accident. The proposed change to the Type A leakage rate testing frequency does not involve any physical changes being made to the facility. In addition, the proposed changes to the Type A leakage rate testing frequency [do] not change the operation of the plants such that a new failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed extension to the Type A testing frequency will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year extension for Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes a very small amount to the individual risk and that the decrease in Type A testing frequency would have a minimal [effect] on this risk since most potential leakage paths are detected by Type B and C testing. The proposed change involves only an extension of the frequency for Type A containment leakage testing; the overall primary containment leakage rate limit specified by Technical Specifications is being maintained. Type B and C containment leakage testing will continue to be performed at the frequency currently required by the BSEP Technical Specifications. The regular containment inspections being performed in accordance with the ASME, Section XI, and the Maintenance Rule (i.e., 10 CFR 50.65) provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing. In addition, the on-line containment monitoring capability that is inherit to [a] boiling water reactor using an inert containment atmosphere allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety is maintained. Therefore, the proposed license amendments do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 5, 2001.

Description of amendment request: The proposed amendment incorporates Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF–364, Revision 0, ''Revision to Technical Specification Bases Control Program to Incorporate Changes to 10 CFR 50.59.'' The proposed change deletes reference to the term ''unreviewed safety question,'' and replaces it with the phrase ''requires NRC approval pursuant to 10 CFR 50.59.''

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 is consistent with the changes described in TSTF change TSTF–364, ''Revision to Technical Specification Bases Control Program to Incorporate Changes to 10 CFR 50.59.'' Specifically, the proposed change deletes the reference to the term ''unreviewed safety question'' as defined in 10 CFR 50.59 (pre-1999 revision) and replaces it with the phrase ''requires NRC approval pursuant to 10 CFR 50.59.'' The deletion of the definition of ''unreviewed safety question'' was approved by the NRC in the current revision of the 10 CFR 50.59 regulation (October 1999). Changes to the Technical Specification Bases will still be evaluated in accordance with the requirements of 10 CFR 50.59. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is to an administrative program. The change does not involve any physical modifications to the facility nor add new equipment. The methods of plant operation have not been altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is administrative in nature, based upon the current version of 10 CFR 50.59 regulation. Changes to the Technical Specification Bases will still be evaluated by 10 CFR 50.59. The proposed change has no direct impact upon any plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 19, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) by the following: (1) Implement programmatic controls for radiological effluent technical specifications in the Administrative Controls section of the TSs, (2) relocate existing procedural details to licensee-controlled documents or new programs to accommodate the incorporation of Generic Letter 89–01 and relevant portions of the Improved Standard Technical Specifications (NUREG–1433), and (3) update the references to current regulatory requirements such as those set forth in 10 CFR 20.1–20.262 and 10 CFR 20.1001–20.2402.

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

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

No. The proposed amendment does not affect accident initiators or precursors because it does not alter any design parameter, condition, equipment configuration, or manner in which the unit is operated. Furthermore, it does not alter or prevent the ability of existing structures, systems, or components to perform their intended safety or accident-mitigating functions depicted in the Updated Final Safety Analysis Report. The proposed amendment is administrative and it only alters the format and location of programmatic controls and procedural details. These changes will not prevent the unit to continue to comply with applicable regulatory requirements. As a result, the proposed amendment will not alter the conditions or assumptions used in previous accident analyses. Therefore, operation in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment does not affect accident initiators or precursors because it does not alter any design parameter, condition, equipment configuration, or manner in which the unit is operated. Furthermore, it does not alter or prevent the ability of structures, systems, or components to perform their intended safety or accident mitigating functions. Accordingly, the proposed amendment does not create any new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment does not change any design parameter, analysis methodology, safety limits or acceptance criteria. It is administrative as outlined above, with the objective to assure continued compliance with applicable regulatory requirements governing the radiation protection plan, radioactive effluents, radioactive sources, and radiological environmental monitoring. Therefore, operation in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: October 26, 2001.

Description of amendment request: Section 6.0 of the Technical Specifications (TSs) delineates the required administrative controls, including plant management responsibilities, station organization, staff qualifications and training, review and audit activities, procedures, reporting requirements, record retention, radiation area control, and various plant programs. The licensee proposed an amendment to revise Section 6.0 of the TSs to make it consistent with its counterpart, Section 5.0, of the Nine Mile Point Nuclear Station, Unit No. 2 (NMP–2) TSs. The NMP–2 TSs was fully converted to the format and style of the Improved Standard Technical Specifications (NUREG–1433 and NUREG–1434) by Amendment No. 91, dated February 15, 2000. While NMP–1 and NMP–2 are of different reactor designs, the administrative controls are, by necessity, either identical or very similar. Consistency of administrative controls between the two units is essential to avoid confusion and to improve efficiency.

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

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

No. The proposed amendment is concerned only with administrative controls, and does not affect accident initiators or precursors because it does not alter any design parameter, condition, equipment configuration, or manner in which NMP–1 is operated. Furthermore, it does not alter or prevent the ability of existing structures, systems, or components to perform their intended safety or accident-mitigating functions depicted in the Updated Final Safety Analysis Report. The proposed amendment only affects the administrative controls in accordance with NUREG–1433 and NUREG–1434. These changes will not prevent NMP–1 to continue to comply with applicable regulatory requirements. As a result, the proposed amendment will not alter the conditions or assumptions used in previous accident analyses. Therefore, operation in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment does not affect accident initiators or precursors because it does not alter any design parameter, condition, equipment configuration, or manner in which the unit is operated. Furthermore, it does not alter or prevent the ability of structures, systems, or components to perform their intended safety or accident mitigating functions. Accordingly, the proposed amendment does not create any new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment does not change any design parameter, analysis methodology, safety limits or acceptance criteria. It is administrative as outlined above, with the objective to assure continued compliance with applicable regulatory requirements governing the various topics of administrative controls. Therefore, operation in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan (Acting).

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

Date of amendment requests: October 17, 2001.

Description of amendment requests: The proposed license amendments would modify Technical Specification 3.9.4, ''Containment Penetrations,'' to allow the equipment hatch, both personnel air lock doors and both emergency air lock doors to remain open, and penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative control, during core alterations and movement of irradiated fuel assemblies. In addition, the amendments revise Technical Specification 1.1, ''Definitions,'' for Dose Equivalent I–131, to allow the use of the thyroid dose conversion factors, listed in the International Commission on Radiological Protection Publication 30, ''Limits for Intakes of Radionuclides by Workers.''

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 would allow the containment equipment hatch, Personnel Air Lock (PAL) doors, Emergency Air Lock (EAL) doors, and penetrations to remain open during fuel movement and core alterations. These penetrations are normally closed during this time period in order to prevent the escape of radioactive material in the event of a Fuel Handling Accident (FHA) inside containment. These penetrations are not initiators of any accident and the probability of a FHA is unaffected by the position of these penetrations.

The new FHA analysis with an open containment demonstrates the maximum offsite doses are well within (less than 25%) the limits specified in 10 CFR 100. These offsite dose values are also well within the acceptable limits provided in NUREG–0800, Section 15.7.4. This FHA analysis results in a maximum offsite dose of 60.62 Rem to the thyroid and 0.4281 Rem to the whole body. The calculated control room dose is also well below the acceptance criteria specified in General Design Criteria (GDC) 19. The analysis results in thyroid and whole body doses to the control room operator of 11.56 Rem and 0.0072 Rem, respectively. Although the offsite and control room dose values are increased by the proposed changes, the resulting values are still well within acceptable limits and do not significantly increase the consequences of a FHA.

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

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

The proposed change does not involve the addition or modification of any plant equipment. However, the proposed change does alter the containment closure configuration and method of operation of the plant during certain operational activities. The proposed change involves a change to the technical specification (TS) that would allow the equipment hatch door, the PAL doors, the EAL doors, and containment penetrations to be open during core alterations and fuel movement inside containment. This change only affects the containment barrier configuration of the plant during certain operational activities. Even allowing these doors and penetrations to be open, all of the resulting radiological consequences remain within acceptable limits and this configuration does not create the possibility of a new or different accident than previously evaluated.

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

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

This proposed change creates the potential for increased dose in the control room and at the site boundary due to a FHA. However, the new analysis demonstrates that the resultant doses are well within the 10 CFR 100 limits and well below the GDC [General Design Criterion] 19 limits. In the case of the offsite dose values, they remain less than 25% of the 10 CFR 100 limits, which is considered acceptable in NUREG–0800 [Standard Review Plan], Section 15.7.4. Based on this, even though the dose values have increased from the previously calculated values, the margin of safety is not significantly reduced.

In the new analysis, the offsite and control room doses due to a FHA with an open containment have been evaluated using conservative assumptions, such as all airborne activity caused by the FHA in the containment is released instantaneously to the outside atmosphere, which ensures the calculation bounds the expected dose. The new analysis also assumes closure of the containment within two hours. As a result, requiring immediate initiation of the closure of the containment and completion of closure within approximately 30 minutes following a containment evacuation requirement from the FHA will reduce the potential offsite doses in the event of a FHA, and provides additional margin to the calculated offsite doses.

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: Christopher J. Warner, 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 (DCPP), Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: November 13, 2001.

Description of amendment requests: The proposed license amendments would modify Technical Specifications 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and 5.6.10, ''SG Tube Inspection Report,'' of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 Technical Specifications, to add new surveillance and reporting requirements associated with SG tube inspection and repair. The new requirements establish alternate repair criteria for axial primary water stress corrosion cracking at dented tube support plate intersections.

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.

Examination of crack morphology for primary water stress corrosion cracking (PWSCC) at dented intersections has been found to show one or two microcracks well aligned with only a few uncorroded ligaments and little or no other inside diameter axial cracking at the intersection. This relatively simple morphology is conducive to obtaining good accuracy in nondestructive examination (NDE) sizing of these indications. Accordingly, alternate repair criteria (ARC) are established based on crack length and average and maximum depth within the thickness of the tube support plate (TSP).

The application of the ARC requires a Monte Carlo condition monitoring assessment to determine the as-found condition of the tubing. The condition monitoring analysis described in WCAP–15573, Revision 1, is consistent with NRC Generic Letter 95–05 [''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking''] requirements.

The application of the ARC requires a Monte Carlo operational assessment to determine the need for tube repair. The repair bases are obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile using Monte Carlo analysis techniques described in WCAP–15573, Revision 1. The burst pressure and leakage are compared to the requirements in WCAP–15573, Revision 1. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected end of cycle (EOC) requirements are satisfied, the tube will be left in service.

A steam generator (SG) tube rupture event is one of a number of design basis accidents that are analyzed as part of a plant's licensing basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. The ARC requires repair of any indication having a maximum crack depth greater than or equal to 40 percent outside the TSP, thus limiting the potential length of a deep crack outside the TSP at EOC conditions and providing margin against burst and leakage for free span indications.

For other design basis accidents such as a MSLB [main steam line break], MFLB [main feed line break], control rod ejection, and locked reactor coolant pump motor, the tubes are assumed to retain their structural integrity.

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.

Implementation of the proposed SG tube ARC does not introduce any significant changes to the plant design basis. A single or multiple tube rupture event would not be expected in a SG in which the ARC has been applied. Both condition monitoring and operational assessments are completed as part of the implementation of ARC to determine that structural and leakage margin exists prior to returning SGs to service following inspections. If the condition monitoring requirements are not satisfied for burst or leakage, the causal factors for EOC indications exceeding the expected values will be evaluated. The methodology and application of this ARC will continue to ensure that tube integrity is maintained during all plant conditions consistent with the requirements of Regulatory Guide (RG) 1.121 [''Bases for Plugging Degraded PWR Steam Generator Tubes''] and Revision 1 of RG 1.83 [Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes].

In the analysis of a SG tube rupture event, a bounding primary-to-secondary leakage rate equal to the operational leakage limits in the Technical Specifications (TS), plus the leak rate associated with the double-ended rupture of a single tube, is assumed. For other design basis accidents, the tubes are assumed to retain their structural integrity and exhibit primary-to-secondary leakage within the limits assumed in the current licensing basis accident analyses. MSLB leakage rates from the proposed PWSCC ARC are combined with leakage rates from other approved ARC (i.e., voltage-based ARC and W* ARC). The combined leakage rates will not exceed the limits assumed in the current licensing basis accident analyses.

The 40 percent maximum depth repair limit for free span indications provides a very low likelihood of free span leakage under design basis or severe accident conditions. Leakage from indications inside the TSP is limited by the constraint of the TSP even under severe accident conditions, and leakage behavior in a severe accident would be similar to that found acceptable by the NRC under approved ARC for axial outside diameter stress corrosion cracking (ODSCC) at TSP intersections. Therefore, even under severe accident conditions, it is concluded that application of the proposed ARC for PWSCC at dented TSP locations results in a negligible difference in risk of a tube rupture or large leakage event, when compared to current 40 percent repair limits or previously approved ARC.

Diablo Canyon Power Plant (DCPP) continues to implement a maximum operating condition leak rate limit of 150 gallons per day per SG to preclude the potential for excessive leakage during all plant conditions.

The possibility of a new or different kind of accident from any previously evaluated is not created because SG tube integrity is maintained by inservice inspection, condition monitoring, operational assessment, tube repair, and primary-to-secondary leakage monitoring.

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

Tube repair limits provide reasonable assurance that tubes accepted for continued service without repair will exhibit adequate tube structural and leakage integrity during subsequent plant operation. The implementation of the proposed ARC is demonstrated to maintain SG tube integrity consistent with the criteria of draft NRC RG 1.121. The guidelines of RG 1.121 describe a method acceptable to the NRC staff for meeting General Design Criteria (GDC) 2, 4, 14, 15, 31, and 32 by ensuring the probability or the consequences of SG tube rupture remain within acceptable limits. This is accomplished by determining the limiting conditions of degradation of SG tubing, for which tubes with unacceptable cracking should be removed from service.

Upon implementation of the proposed ARC, even under the worst-case conditions, the occurrence of PWSCC at the tube support plate elevations is not expected to lead to a SG tube rupture event during normal or faulted plant conditions. The ARC involves a computational assessment to be completed for each indication left in service ensuring that performance criteria for tube integrity and leak tightness are met until the next scheduled outage.

As discussed below, certain tubes are excluded from application of ARC. Existing tube integrity requirements apply to these tubes, and the margin of safety is not reduced. In addressing the combined loading effects of a loss-of-coolant (LOCA) and safe shutdown earthquake (SSE) on the SGs (as required by GDC 2), the potential exists for yielding of the TSP in the vicinity of the wedge groups, accompanied by deformation of tubes and a subsequent postulated in-leakage. Tube deformation could lead to opening of pre-existing tight through wall cracks, resulting in secondary to primary in-leakage following the event, which could have an adverse affect on the Final Safety Analysis Report (FSAR) results. Based on a DCPP analysis of LOCA and SSE, SG tubes located in wedge region exclusion zones are susceptible to deformation, and are excluded from application of ARC.

A DCPP tube stress analysis for MFLB/MSLB plus SSE loading determined that high bending stresses occur in certain SG tubes at the seventh TSP, because the stresses exceed the maximum imposed bending stress for existing test data (equal to approximately the lower tolerance limit yield stress). These tubes are located in rows 11 to 15 and 36 to 46, and are excluded from application of ARC.

Tube intersections that contain TSP ligament cracking are also excluded from application of ARC.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to the plant safety analyses as defined in the FSAR or TS.

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

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

Attorney for licensee: Christopher J. Warner, 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 (DCPP), Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: November 16, 2001.

Description of amendment requests: The proposed license amendments would modify Technical Specification 5.5.16, ''Containment Leakage Rate Testing Program,'' to allow a one-time extension of the ten-year interval for the performance-based leakage rate testing program for Type A tests as prescribed by Nuclear Energy Institute (NEI) Report NEI 94–01, Revision 0, ''Industry

Guideline for Implementing Performance-Based Option of 10 CFR part 50, Appendix J,'' and applied by 10 CFR part 50, Appendix J, Option 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed extension to the Type A testing interval from 1-in-10 years to 1-in-15 years will not increase the probability of an accident previously evaluated. The determination of containment integrity is not an accident initiator. The containment Type A testing interval extension does not involve a plant modification and the testing interval extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. Research documented in NUREG–1493 has determined that Type B and C tests can identify the vast majority (approximately 97 percent) of all potential leakage paths. Experience at Diablo Canyon Power Plant (DCPP) demonstrates that excessive containment leakage paths are detected by Type B and C local leakage rate tests. Type B and C testing will identify any containment opening, such as a valve, that would otherwise be detected by the Type A tests.

NUREG–1493 concluded that increasing the Type A test interval to 1-in-20 years leads to an imperceptible increase in risk. A DCPP plant specific probabilistic risk assessment of the change in the Type A testing interval from 1-in-10 years to 1-in-15 years determined the total integrated risk of the associated specific accident sequences increases by 0.03 percent. This risk impact when compared to other severe accident induced risks is negligible. The increase in the Regulatory Guide (RG) 1.174 [An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis] large early release fraction (LERF) figure-of-merit criteria resulting from a change in the Type A test interval from 1-in-10 years to 1-in-15 years is risk insignificant.

Testing and inspection provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The structural capability of the containment has been shown by the Type A testing results that have established that DCPP has had acceptable containment leakage rates with considerable margin. Inspections required by 10 CFR 50.65 and American Society of Mechanical Engineers code are performed in order to identify indications of containment degradation that could affect leak tightness. The results of containment concrete examination have concluded the containment concrete has had no loss of structural capacity and no areas of the concrete shell have experienced accelerated degradation or aging. The results of containment liner inspections have not identified any significant degradations that could adversely impact the containment structural integrity or leak tightness, such as through-holes in the containment liner. Due to the large containment leakage rate margin available, and no identified mechanism that would cause significant degradation of containment, a 5 year extension of the ILRT interval would not be expected to result in containment leakage above the acceptable limit.

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 extension of the Type A testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no physical changes being made to the plant and there are no changes in operation of the plant that could introduce a new failure mode, creating an accident.

The containment structure is passive. Under normal operating conditions, there is no significant environmental or operational stress present that would contribute to its degradation. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests.

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

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

The proposed extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing results in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the increase in the Type A testing interval would have a minimal effect on this risk because 97 percent of the potential leakage paths are detected by Type B and C testing.

A DCPP plant specific probabilistic risk assessment of the change in the Type A testing interval from 1-in-10 years to 1-in-15 years determined the total integrated risk of the associated specific accident sequences increases by 0.03 percent. This risk impact when compared to other severe accident induced risks is negligible. The increase in RG 1.174 LERF figure-of-merit criteria resulting from a change in the Type A test interval from 1-in-10 years to 1-in-15 years is risk insignificant.

Deferral of Type A testing for DCPP does not increase the level of risk to the public due to loss of capability to detect and measure containment leakage or loss of containment structural capability. Other containment testing methods and inspections will assure all limiting conditions of operation will continue to be met. The margin of safety inherent in existing accident analyses is maintained.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: November 16, 2001.

Description of amendment requests: The proposed license amendments would modify Technical Specification (TS) 1.1, ''Definitions, Dose Equivalent I–131,'' to allow the use of the thyroid dose conversion factors listed in the International Commission on Radiological Protection (ICRP) Publication 30, ''Limits for Intakes of Radionuclides by Workers,'' 1979, in the steam generator tube rupture and main steam line break radiological consequences analyses.

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 revision of Technical Specification (TS) 1.1, Definitions, ''Dose Equivalent I–131,'' to allow use of the iodine thyroid dose conversion factors from the International Commission on Radiological Protection (ICRP) Publication 30, 1979, and the revised steam generator tube rupture (SGTR) and main steam line break (MSLB) radiological consequences analyses are used to determine post-accident dose. They are not related to any accident initiator. Therefore, this change cannot increase the probability of an accident.

The revised SGTR thermal and hydraulic analysis input assumptions are consistent with actual plant limits and parameters.

The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR Part 100 limits and the NUREG–0800 Standard Review Plan (SRP) section 15.1.5 and section 6.4 guideline values.

The revised SGTR control room radiological consequences analysis dose results are within the SRP section 6.4 guideline values.

The revised SGTR offsite radiological consequences analysis dose results are within the 10 CFR part 100 dose limits. The SGTR offsite dose results also meet the SRP section 15.6.3 and section 6.4 guideline values, with the exception of the 2 hour Exclusion Area Boundary (EAB) thyroid dose. The calculated 2 hour EAB thyroid dose of 30.5 Rem is 1.5 percent above the SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB thyroid dose has been compared against the conservative thyroid dose SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB dose thyroid dose would be equivalent to a Regulatory Guide (RG) 1.183 methodology Total Effective Dose Equivalent (TEDE) of approximately 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem is not considered to be a significant increase in dose.

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 use of the iodine thyroid dose conversion factors from ICRP Publication 30 and the revised SGTR and main steam line break MSLB radiological consequences analyses do not involve any physical plant changes. The change does not involve changes in operation of the plant that could introduce a new failure mode for creating an accident or affect the mitigation of an accident.

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

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

The use of ICRP Publication 30 thyroid dose conversion factors to calculate the radiological consequences for a SGTR and MSLB accident is endorsed by RG 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' US Nuclear Regulatory Commission, July 2000. Therefore, the revision of TS 1.1, Definitions, ''Dose Equivalent I–131,'' to allow use of the iodine thyroid dose conversion factors from ICRP Publication 30 does not result in a significant reduction in the margin provided by TS 1.1. The revised SGTR thermal and hydraulic analysis input assumptions are consistent with actual plant limits and parameters.

The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR part 100 limits and the NUREG–0800 SRP section 15.1.5 and section 6.4 guideline values.

The revised SGTR control room radiological consequences analysis dose results are within the SRP section 6.4 guideline values.

The revised SGTR radiological consequences analysis dose results are within the 10 CFR part 100 dose limits. The SGTR dose results also meet the SRP section 15.6.3 and section 6.4 guideline values, with the exception of the 2 hour EAB thyroid dose. The calculated 2 hour EAB thyroid dose of 30.5 Rem is 1.5 percent above the SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB dose thyroid dose would be equivalent to a RG 1.183 methodology TEDE of approximately 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem is not a significant reduction in the margin of safety.

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

NRC Section Chief: Stephen Dembek.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: December 11, 2001.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' on the emergency diesel generators. The revisions would change (1) Conditions A and C of the Actions for Limiting Condition for Operation 3.8.3, and (2) Surveillance Requirement 3.8.3.1. The proposed amendments would change the minimum required diesel fuel oil storage to support (1) using California Diesel fuel rather than the existing Environmental Protection Agency (EPA) Clear diesel fuel, (2) revising the diesel generator load profile in reactor Modes 1 through 4, and (3) changing the units of the required diesel fuel oil storage. A ''greater than or equal to'' would also be changed to a ''greater than'' sign.

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.

This proposed change revises the minimum amount of stored diesel fuel. The change is required to (1) support the use of California Diesel fuel rather than the existing EPA Clear diesel fuel, and (2) reflect a change in the diesel generator load profile in Modes 1 through 4.

In addition, this proposed change revises the units for the minimum diesel fuel storage requirements from tank level to a minimum required volume of fuel in gallons. A ''greater than or equal to'' sign is revised to a ''greater than'' sign for consistency. These are administrative changes only.

Technical Specification (TS) 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air,'' requires that each diesel generator have sufficient fuel to operate for a period of 7 days, while the Diesel Generator (DG) is supplying maximum post Loss of Coolant Accident (LOCA) load demand. This requirement is currently expressed as a minimum tank level limit. In Modes 1 through 4, the existing tank level limit is 89%, which ensures that a 7-day supply of fuel is available. TS 3.8.3, Condition A, states that during Modes 1 through 4, if one or more Diesel Generators (DG) has a fuel level in the storage tank less than 89% and greater than or equal to 76%, then fuel oil level must be restored to within limits within 48 hours. The 76% level requirement is based on maintaining a 6-day supply of diesel fuel in Modes 1 through 4. If the tank level is at or below 76% (6-day supply), the associated DG must be declared inoperable immediately.

Similarity, for Modes 5 and 6, the existing tank level limit is 72%, which ensures that a 7-day supply of fuel is available. TS 3.8.3, Condition C, states that during Modes 5 and 6, if one required DG has a fuel level in the storage tank less than 72% and greater than 63%, then [the] fuel oil level must be restored to within limits within 48 hours. The 63% level requirement is based on maintaining a 6-day supply of diesel fuel in Modes 5 and 6. If the tank level is at or below 63% (6-day supply), the associated diesel generator must be declared inoperable immediately.

As described in the Bases to TS 3.8.3, these tank level requirements are based on fuel volume requirements. In Modes 1 through 4, 89% and 76% level limits are based on a 7-day (49,724 gallons) and 6-day (42,960 gallons) fuel supply, respectively [plus an allowance for instrument Total Loop Uncertainty (TLU)]. In Modes 5 and 6, the 72% and 63% tank level limits are based on a 7-day (40,472 gallons) and 6-day (34,960 gallons) fuel supply, respectively (plus an allowance for instrument TLU).

Because the Lower Heating Value (LHV) per gallon of California Diesel fuel is less than that of EPA Clear diesel fuel, it was necessary to recalculate the amount of fuel required to supply necessary loads for the required time periods. For Modes 1 through 4, the resulting minimum volumes of California Diesel fuel are 45,662 gallons and 39,468 gallons for the 7-day and 6-day fuel supply, respectively. For Modes 5 and 6, the required volumes of California Diesel fuel are

41,691 gallons and 35,735 gallons for a 7-day supply and a 6-day supply, respectively.

It should be noted that the minimum volumes for Modes 1 through 4 are decreased due to a change in the calculated [diesel generator] load profile. SONGS no longer requires the third-of-a-kind High Pressure Safety Injection (HPSI) pump to be started following a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). Operation of the third-of-a-kind HPSI pump is no longer assumed as part of the DG load profile in Modes 1 through 4. This resulted in a net decrease in the amount of required stored diesel fuel in Modes 1 through 4, even when the use of the California Diesel fuel [with the lower heating value] is taken into account.

The diesel generators and the associated support systems such as the fuel oil storage and transfer systems are designed to mitigate accidents and are not accident initiators. Revising the minimum volumes of stored [diesel] fuel in the storage tanks will not result in a significant increase in the probability of any accident previously evaluated. [The revisions are to maintain the current requirements for a 7-day and 6-day supply of stored diesel fuel].

Following implementation of this proposed change, there will be no change in the ability of the diesel generators to supply maximum post-LOCA load demand for 7 days. The proposed minimum volumes of fuel, 45,662 gallons and 39,468 gallons, ensure that a 7-day and 6-day supply of fuel, respectively, are available in Modes 1 through 4. The proposed minimum volumes of fuel, 41,691 gallons and 35,735 gallons, ensure that a 7-day and a 6-day supply, respectively, of fuel is available in Modes 5 and 6. This is identical to the current requirements. Therefore this change will not result in a significant increase in the consequences of any accident previously evaluated.

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

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

Response: No.

Following this change, the diesel generators will still be able to supply maximum post-LOCA load demand. The current 7-day and 6-day fuel supply requirements will be maintained following this change. [The diesel generators fuel oil storage and transfer systems are not accident initiators].

The changes in units from tank level percentage to fuel volume in gallons is an administrative change only. The change from a ''greater than or equal to'' sign to a ''greater than'' sign in TS 3.8.3, Condition A, is for consistency with other parts of TS 3.8.3 and is also an administrative change.

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

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

Response: No.

The Bases to TS 3.8.3 states that ''Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days, while the DG is supplying maximum post loss of coolant accident load demand.'' When the fuel oil tank level is less than required to support [7 days] of operation, the required action depends on whether or not a 6-day supply of fuel is available. [The proposed tank level limits for Modes 1 through 4 will maintain the 7-day and 6-day fuel supply requirements following changeout to California Diesel fuel and the change in the DG load profile for Modes 1 through 4].

The proposed tank level limits for Modes 5 and 6 will maintain these 7-day and 6-day fuel supply requirements following changeout to California Diesel fuel.

The change in units from tank level percentage to fuel volume in gallons is an administrative change only. The change from a ''greater than or equal to'' sign to a ''greater than'' sign in TS 3.8.3, Condition A, is for consistency with other parts of the TS 3.8.3 and is also an administrative change.

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: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of amendment request: November 1, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 2.1.1.2, ''Reactor Core Safety Limits,'' by modifying the safety limit minimum critical power ratio (SLMCPR).

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:

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

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

The proposed amendment establishes revised SLMCPR values for two recirculation loop operation and for single recirculation loop operation. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Since the change does not require any physical plant modifications or physically affect any plant components, no individual precursors of an accident are affected and the probability of an evaluated accident is not increased by revising the SLMCPR values.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The revised SLMCPRs have been performed using NRC-approved methods and procedures. The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. These calculations do not change the method of operating the plant and have no effect on the consequences of an evaluated accident. Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed license amendment involves a revision of the SLMCPR for two recirculation loop operation and for single loop operation based on the results of an analysis of the Cycle 11 core. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in the allowable methods of operating the facility. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable methods of operation. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in the TS bases will remain the same. The new SLMCPRs are calculated using NRC-approved methods and procedures which are in accordance with the current fuel design and licensing criteria. The SLMCPRs remain high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of application for amendments: September 12, 2001 (TS 01–04).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Unit 1 and 2 Technical Specification (TS) 3/4 6.5.1 and associated Bases to reflect an increase in the ice condenser basket weight from 1071 pounds to 1145 pounds and the total ice condenser ice weight from 2,082,024 pounds to 2,225,880 pounds. This change is being made in response to a reanalysis by Westinghouse Electric Company that identified a modeling input error used in the original analysis.

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

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

The analyzed accidents of consideration in regards to changes affecting the ice condenser are a loss-of-coolant accident (LOCA) and a main steam line break (MSLB) inside containment. The ice condenser is a passive system and is not postulated as being the initiator of any LOCA or main steam line break (MSLB) and is designed to remain functional following a design basis earthquake.

In addition, the ice condenser does not interconnect or interact with any systems that have an interface with the reactor coolant or main steam systems.

For SQN, the LOCA is the more severe accident in terms of containment pressure and ice bed meltout and is therefore the more limiting accident. SQN's LOCA Containment Integrity Analysis calculates the post-LOCA peak containment pressure to be 11.44 pounds per square inch gauge (psig), which is below SQN's containment design pressure of 12.0 psig. The analysis contains an assumed ice mass that is an input value to the calculation to ensure that sufficient heat removal capability is available from SQN's ice condenser to limit the accident peak pressure inside containment. The analyzed peak accident pressure must remain below the containment design pressure.

TVA's proposed TS revision reflects the ice mass assumed in the SQN ['s] Containment Integrity Analysis. Accordingly, TVA's proposed change ensures that ice mass values retain the existing margin between the calculated peak containment accident pressure and SQN's containment design pressure.

Since the proposed changes to the TS and TS bases are solely to revise ice weight values to reflect current margins within SQN's analysis, and are not the result of or require any physical changes to the ice condenser, there is no change in the probability of an accident previously evaluated in the Safety Analysis Report.

Based on the above discussions, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Because the TS and TS bases changes do not involve any physical changes to the ice condenser, or chemical changes to the ice contained therein, or make any changes in the operational aspects of the ice condenser as required by the TS, there are no new or different kind of accidents created from those already identified and evaluated.

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

The ice condenser TSs ensure that during a LOCA or MSLB the ice condenser will initially pass sufficient air and steam mass to preclude over-pressurizing lower containment, that it will absorb sufficient heat energy initially and over a prescribed time period to assist in precluding containment vessel failure, and that it will not alter the bulk containment sump pH and boron concentration assumed in the accident analysis.

TVA's proposed change does not physically alter the ice condenser, but rather accounts for changes to input assumptions for SQN's containment pressure analysis to correct a computer model input error. The correction to the model provides a more accurate accounting of the pressure response inside containment following a LOCA. The error correction requires an increase the ice mass assumed in the analysis to ensure that SQN's post-LOCA peak containment pressures remain unchanged. The margin that exists between the accident peak pressure and the containment design pressure is unaffected. Accordingly, TVA' s proposed change does not reduce the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: November 20, 2001.

Description of amendment request: The proposed amendment would change the Technical Specifications Table 3.2.6 by revising the Allowed Outage Times (AOTs) and associated action requirements for certain post-accident monitoring (PAM) instrumentation.

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

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

The PAM instrumentation is not considered as an initiator or contributor to any previously evaluated accident. The proposed change will not affect any Final Safety Analysis Report safety analysis. Because there are no credible failures of the PAM instrumentation that could initiate any accidents previously evaluated, changing the AOTs and related actions for PAM instrumentation will not increase the probability of any accident previously evaluated. The operability or inoperability of this instrumentation will not cause an accident because this instrumentation was not intended to and does not serve a function for preventing accidents. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The availability and use of PAM instrumentation ensures that the prescribed operator (manual) actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred as intended. The availability and use of PAM instrumentation provide assurance that the consequences of accidents will not be greater than previously evaluated. Changes to allowed outage times and shutdown completion times do not affect the consequences of accidents. The proposed change does not modify any parameters or assumptions contained in previously analyzed design-basis events. The continued availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred as intended. The requirements of the revised TS are adequate to ensure the required instrumentation is maintained operable such that PAM instrumentation will be available to perform its intended safety function. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve any physical modification to the plant, change in TS setpoints, plant design-basis, or the manner in which the plant is operated. Because the PAM instrumentation serves a passive function and does not provide any automatic action, there are no credible failures of the PAM instrumentation that could initiate a new or different kind of accident from any accident previously evaluated.

The change to AOTs and related action requirements for PAM instrumentation will not result in a failure mode not previously analyzed. This instrumentation is not considered an accident precursor because its existence or availability does not have any adverse impact in the pre-accident state of the reactor.

Therefore, this change does 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?

PAM instrumentation is assumed to be used by operators for monitoring only after an accident occurs and performs no automatic functions. The continued availability and use of this instrumentation ensures that the prescribed manual operator actions for mitigating the consequences of an accident will be implemented when necessary, and that the operator has sufficient information to verify required automatic actions have occurred as intended. The requirements of the revised TS are adequate to ensure the required instrumentation is maintained operable such that PAM instrumentation will be available to perform its intended safety function.

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

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

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

NRC Section Chief: James W. Clifford.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: December 11, 2001.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

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

The licensee affirmed the applicability of the following NSHC determination in its application dated December 11, 2001.

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in a margin of safety. Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor) Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Document 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.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC Public Document Room Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by e-mail to *pdc@nrc.gov.*

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

Date of application for amendment: December 29, 2000, supplemented on October 11, 2001.

Brief description of amendment: The amendment revises the offsite power source identified in Technical Specification 3.7.A.3 to remove one listed source and add a different source. Also, the bases have been revised to reflect the availability of the offsite sources and to revise minor administrative changes.

Date of Issuance: December 27, 2001.

Effective date: December 27, 2001, and shall be implemented within 30 days of issuance.

Amendment No.: 222.

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

*Date of initial notice in **Federal Register:*** April 4, 2001 (66 FR 17965).

The October 11, 2001, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated December 27, 2001.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: April 11, 2001, as supplemented June 14, 2001.

Brief description of amendment: The amendment revises the list of documents that describe the analytical methods used to determine the core operating limits specified in Technical Specification 6.9.1.8b. The revision consists of updating the list of documents to include the latest NRC-approved methodologies, along with deleting the revision numbers and dates of all documents in the list.

Date of issuance: December 19, 2001.

Effective date: As of the date of issuance and shall be implemented prior to restart from refueling outage 14 which is currently scheduled in early February of 2002.

Amendment No.: 260.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38760).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 28, 2001.

Brief description of amendment: The amendment deletes the post-maintenance testing Surveillance Requirement 4.6.3.1 of containment isolation valves.

Date of issuance: December 21, 2001.

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

Amendment No.: 200.

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

*Date of initial notice in **Federal Register:*** October 17, 2001 (66 FR 52799).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 2001.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: June 12, 2001, as supplemented by letters dated October 15 and November 16, 2001.

Brief description of amendment: The amendment revised certain emergency diesel generator (EDG) Technical Specifications (TSs) to remove the requirement for an accelerated test frequency, remove the requirement to subject the EDGs to an inspection in accordance with the manufacturer's recommendations, allow that certain EDG tests may be done in modes other than shutdown, and remove the EDG special reporting requirements.

Date of issuance: December 17, 2001.

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

Amendment No.: 237.

Facility Operating License No. NPF–6: Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** July 11, 2001 (66 FR 36340). The October 15 and November 16, 2001, supplemental letters provided clarifying information and revised TSs that were within the scope of the original **Federal**

Register notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 2001.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 22, as supplemented by letters dated May 4, and September 13, 2001.

Brief description of amendments: The amendments revise reactor vessel water level—low scram and isolation setpoints in order to minimize unnecessary reactor scrams that might result from events involving a temporary reduction in feedwater flow. The revision to these setpoints was originally requested as part of the power uprate licensing amendment. However, since the setpoint reduction will provide a similar benefit when operating at the current thermal power, Exelon has requested to implement the requested changes prior to power uprate approval as part of efforts to improve summer reliability at Dresden and Quad Cities Nuclear Power Stations.

Date of issuance: December 18, 2001.

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

Amendment Nos.: Dresden Units 2 and 3—190/184; Quad Cities Units 1 and 2—200/196.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 6, 2001 (66 FR 30490).

The supplemental letters 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 amendments is contained in a Safety Evaluation dated December 18, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 29, 2000, as supplemented by letters dated March 1, July 13, August 9, August 13, August 27, and October 17, 2001.

Brief description of amendments: The amendments change the technical specifications to reflect a change in fuel vendors from Siemens Power Corporation to General Electric, and a transition to GE14 fuel. As part of the transition, changes are made to (1) increase the number of required automatic depressurization system valves from four to five, and (2) remove an allowance to continue operating for 72 hours if certain combinations of emergency core cooling systems are inoperable.

Date of issuance: December 20, 2001.

Effective date: As of the date of issuance and shall be implemented as follows: for Unit 1, prior to reaching Startup (i.e., Mode 2) following refueling outage 17, scheduled for completion in November 2002; for Unit 2, prior to reaching Startup (i.e., Mode 2) following refueling outage 16, scheduled for completion in February 2002.

Amendment Nos.: 201 and 197.

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

Date of initial notice in ***Federal Register:*** December 27, 2000, (65 FR 81912) and August 22, 2001 (66 FR 44172).

The submittals dated July 13, August 9, August 13, August 27, and October 17, 2001, did not change the scope of the amendment or the proposed no significant hazards findings dated December 27, 2000, (65 FR 81912) and August 22, 2001 (66 FR 44172).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 17, 2001.

Brief description of amendments: Revised St. Lucie Unit 1 and 2 Technical Specifications (TS) actions regarding inoperable redundant components when an Emergency Diesel Generator becomes inoperable, such that required actions will be based on the TS for the inoperable redundant components.

Date of Issuance: December 17, 2001.

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

Amendment Nos.: 180 and 123.

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

Date of initial notice in ***Federal Register:*** November 14, 2001 (66 FR 57121).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 2001.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: October 23, 2000.

Brief description of amendments: The amendments removed references to the containment hydrogen monitors in Technical Specification (TS) Tables 3.3–5, ''Accident Monitoring Instrumentation,'' and 4.3–4, ''Accident Monitoring Instrumentation Surveillance Requirements.'' In addition, the amendments deleted TS 3/4.6.5, ''Combustible Gas Control—Hydrogen Monitors,'' and TS 3/4.6.6, ''Post Accident Containment Vent System.''

Date of issuance: December 20, 2001.

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

Amendment Nos: 217 and 211.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2014).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: October 26, 2001.

Brief description of amendment: Amendment changes the Operating License to extend certain Technical Specification surveillance requirement intervals on a one-time basis.

Date of issuance: December 19, 2001.

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

Amendment No.: 206.

Facility Operating License No. DPR–20. Amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** November 13, 2001 (66 FR 56865).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 2001.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: September 19, 2001, as supplemented October 26, 2001.

Brief description of amendment: The amendment authorized a one-cycle delay in removal of the second reactor pressure vessel material surveillance capsule.

Date of issuance: December 20, 2001.

Effective date: As of date of issuance.

Amendment No.: 197.

Facility Operating License No. NPF–14: The amendment authorized a change to the Reactor Vessel Material Surveillance Program.

Date of initial notice in ***Federal Register:*** November 14, 2001 (66 FR 57122).

The October 26, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of application for amendments: December 10, 2001, as supplemented on December 21, and December 24, 2001.

Brief description of amendments: The amendment allows a one-time change to Technical Specifications (TSs) Limiting Condition for Operation 3/4.7.4, ''Service Water System,'' to increase the Allowed Outage Time (AOT) from 72 hours to 10 days. The increase in the TS 3/4.7.4 AOT is necessary in order to allow repairs to a portion of the 12 Service Water System piping while remaining at power.

Date of issuance: December 27, 2001.

Effective date: As of the date of issuance.

Amendment No.: 248.

Facility Operating License Nos. DPR–70: The amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration determination are contained in a Safety Evaluation dated December 27, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 5, 2001.

Brief description of amendments: The amendments revise Technical Specification 3/4.6.4, ''Containment Systems, Combustible Gas Control, Hydrogen Analyzers,'' to reduce the channel calibration frequency of the Hydrogen Analyzers from quarterly to a frequency of once per refueling outage. The change also adds an additional surveillance requirement to perform a quarterly gas calibration.

Date of issuance: December 17, 2001.

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

Amendment Nos.: 247 and 228.

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

Date of *initial notice in* ***Federal Register:*** April 18, 2001 (66 FR 20008).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 2001.

No significant hazards consideration comments received: No.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: October 1, 2001.

Brief description of amendment: This amendment deletes Technical Specification 6.8.4.d requiring a program for post-accident sampling, and thereby eliminates the requirements to have and maintain the Post-Accident Sampling System at Virgil C. Summer Nuclear Station, Unit No. 1.

Date of issuance: December 20, 2001.

Effective date: December 20, 2001.

Amendment No.: 152.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 31, 2001 (66 FR 55023).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 2001.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: December 8, 2000.

Brief description of amendments: The amendments delete or modify existing license conditions from the Unit 1 and Unit 2 Operating License, which have been completed or are otherwise no longer in effect. These activities have now been completed and the license conditions are either obsolete or are no longer needed.

Date of issuance: December 7, 2001.

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

Amendment Nos.: 152 and 144.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Operating License.

Date of initial notice in ***Federal Register:*** March 7, 2001 (66 FR 13808).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: September 10, 2001.

Brief description of amendment: These amendments eliminate the Technical Specification requirements to have and maintain a post-accident sampling system (PASS) at North Anna Power Station, Units 1 and 2. In addition, for North Anna Power Station, Unit 2, the amendment deletes a license condition associated with the implementation of PASS.

Date of issuance: December 19, 2001.

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

Amendment Nos.: 229 and 210.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the Technical Specifications for both units and the license for Unit 2 only.

*Date of initial notice in **Federal Register:*** October 31, 2001 (66 FR 55025).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 10, 2001.

Brief Description of amendments: These amendments eliminate the Technical Specification requirements to have and maintain a post-accident sampling system at Surry Power Station, Unit Nos. 1 and 2.

Date of issuance: December 18, 2001.

Effective date: December 18, 2001.

Amendment Nos.: 229 and 229.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 31, 2001 (66 FR 55026).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 18, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 31st day of December 2001.

For the Nuclear Regulatory Commission.

Stuart A. Richards,

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

[FR Doc. 02–301 Filed 1–7–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued a revision of a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

Revision 1 of Regulatory Guide 1.78, ''Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,'' describes guidance acceptable to the NRC staff for assessing the habitability of the control room during and after a postulated external release of hazardous chemicals. This guide also provides guidance for the protection of control room operators against an accidental release of hazardous chemicals, including chlorine.

With the publication of Regulatory Guide 1.78, Regulatory Guide 1.95, ''Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release,'' is being withdrawn because the guidance in Regulatory Guide 1.95 has been updated and incorporated into Revision 1 of Regulatory Guide 1.78.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Regulatory guides are available for inspection or downloading at the NRC's Web site at *www.nrc.gov* under Regulatory Guides and in NRC's Electronic Reading Room (ADAMS System) at the same site. Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301) 415–2289, or by E-mail to *distribution@nrc.gov.* Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 26th day of December, 2001.

For the Nuclear Regulatory Commission.

Michael E. Mayfield,

Director, Division of Engineering Technology, Office of Nuclear Regulatory Research.

[FR Doc. 02–406 Filed 1–7–02; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

United States Postal Service Board of Governors; Sunshine Act Meeting

Board Votes To Close December 24, 2001, Meeting

By paper and telephone vote on December 21 and 24, 2001, a majority of the members contacted and voting, the Board of Governors of the United States Postal Service voted to close to public observation its meeting held in Washington, DC via teleconference. The Board determined that prior public notice was possible.

Item Considered: Rate Case R2001–1.

General Counsel Certification: The General Counsel of the United States Postal Service has certified that the meeting was properly closed under the Government in the Sunshine Act.

Contact Person for More Information: Requests for information about the meeting should be addressed to the Secretary of the Board, David G. Hunter, at (202) 268–4800.

David G. Hunter,

Secretary.

[FR Doc. 02–512 Filed 1–4–02; 12:20 pm]

BILLING CODE 7710–12–M

RAILROAD RETIREMENT BOARD

Determination of Quarterly Rate of Excise Tax for Railroad Retirement Supplemental Annuity Program

In accordance with directions in Section 3221(c) of the Railroad Retirement Tax Act (26 U.S.C., Section 3221(c)), the Railroad Retirement Board has determined that the excise tax imposed by such Section 3221(c) on every employer, with respect to having individuals in his employ, for each work-hour for which compensation is paid by such employer for services rendered to him during the quarter beginning January 1, 2002, shall be at the rate of 25 cents.

In accordance with directions in Section 15(a) of the Railroad Retirement Act of 1974, the Railroad Retirement Board has determined that for the quarter beginning January 1, 2002, 41.1 percent of the taxes collected under Sections 3211(b) and 3221(c) of the Railroad Retirement Tax Act shall be credited to the Railroad Retirement Account and 58.9 percent of the taxes collected under such Sections 3211(b) and 3221(c) plus 100 percent of the taxes collected under Section 3221(d) of the Railroad Retirement Tax Act shall be credited to the Railroad Retirement Supplemental Account.