INDIVIDUALS RECEIVING ADVANCE NOTIFICATION OF NUCLEAR WASTE SHIPMENTS—Continued

State	Part 71	Part 73
Virginia	Brett A. Burdick, Director, Technological Hazards Division, Department of Emergency Management, Commonwealth of Virginia, 10501 Trade Court, Richmond, VA 23236, (804) 897–6500, ext. 6569, 24 hours: (804) 674–2400.	Same.
Washington	Steven L. Kalmbach, Assistant State Fire Marshall, Washington State Patrol, Fire Protection Bureau, P.O. Box 42600, Olympia, WA 98504–2600, (360) 570–3119, 24 hours: (1–800) 409–4755.	Same.
West Virginia	Colonel H. E. Hill, Jr., Superintendent, West Virginia State Police, 725 Jefferson Road, South Charleston, WV 25309, (304) 746–2111.	Same.
Wisconsin	Edward J. Gleason, Administrator, Division of Emergency Management, 2400 Wright Street, P.O. Box 7865, Madison, WI 53707–7865, (608) 242–3232.	Same.
Wyoming	Captain Vernon Poage, Support Services Officer, Commercial Carrier, Wyoming Highway Patrol, 5300 Bishop Boulevard, Cheyenne, WY 82009–3340, (307) 777–4317, 24 hours: (307) 777–4321.	Same.
District of Columbia	Gregory B. Talley, Program Manager, Radiation Protection Division, Bureau of Food, Drug & Radiation Protection, Department of Health, 51 N Street, NE., Room 6006, Washington, DC 20002, (202) 535–2320, 24 hours: (202) 666–8001.	Same.
Puerto Rico	Esteban Mujica, Chairman, Environmental Quality Board, P.O. Box 11488, San Juan, PR 00910, (787) 767–8056 or (787) 767–8181.	Same.
Guam	Jesus T. Salas, Administrator, Guam Environmental Protection Agency, P.O. Box 22439 GMF, Barrigada, Guam 96921, (671) 457–1658.	Same.
Virgin Islands	Dean C. Plaskett, Esq., Commissioner, Department of Planning and Natural Resources, Cyril E. King Airport, Terminal Building—Second Floor, St. Thomas, Virgin Islands 00802, (340) 774–3320.	Same.
American Samoa	Pati Faiai, Government Ecologist, Environmental Protection Agency, Office of the Governor, Pago Pago, American Samoa 96799, (684) 633–2304.	Same.
Commonwealth of the Northern Mariana Islands.	Thomas B. Pangelinan, Secretary, Department of Lands and Natural Resources, Commonwealth of Northern Mariana Islands Government, Caller Box 10007, Saipan, MP 96950, (670) 322–9830 or (670) 322–9834.	Same.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, June 13, 2003, through June 26, 2003. The last biweekly notice was published on June 24, 2003 (68 FR 37574).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the

Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By August 7, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the

Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

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

Date of amendment request: June 2, 2003.

Description of amendment request: The licensee proposed to revise Sections 3.7.B.1 and 3.7.C.2 of the OCNGS Technical Specifications (TSs). Section 3.7.B.1 currently specifies that the reactor may remain in operation ''for a period not to exceed 7 days in any 30 day period if a startup transformer is out of service.'' Section 3.7.C.2, referring to the standby diesel generators (DGs), currently specifies that the reactor may remain in operation ''for a period not to exceed 7 days in any 30 day period if a diesel generator is out of service.'' The proposed revision is to delete the phrase ''in any 30 day period'' from these two sections. The licensee regards this phrase as an unnecessary restriction, and states that it has no basis in the existing TSs, design basis, or licensing basis of OCNGS.

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes, if approved by the NRC staff, will be made in a manner such that conservatism is maintained through continued compliance with applicable NRC regulations (specifically, the Maintenance Rule in 10 CFR 50.65) and guidance. No hardware design change is involved with the proposed amendment, thus there can be no adverse effect on the functional performance of the startup transformers or DGs. Consequently, the subject components will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. Unavailability of these components was not factored into the scenarios of previously analyzed accidents, nor were the subject components assumed to be initiators of previously analyzed accidents. Consequently, the proposed revision to the subject sections will lead to no increase in the consequences of accidents previously evaluated, and will lead to no increase of the probability of accidents previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods OCNGS is operated. As a result, all structures, systems, and components will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the licensee did not propose to exceed or alter a design basis or safety limit, the proposed amendment will not affect in any way the performance characteristics and intended functions of the subject components. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John E. Matthews, Esquire, Morgan, Lewis, & Bockius, LLP, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Richard J. Laufer.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of amendments request: May 28, 2003.

Description of amendments request: The amendments would modify several surveillance requirements (SRs) in Technical Specifications (TSs) 3.8.1 and 3.8.4 on alternating current and direct current sources, respectively, for plant operation. The revised SRs would have notes deleted or modified to allow the SRs to be performed, or partially performed, in reactor modes that are currently not allowed by the TSs. The current SRs are not allowed to be performed in Modes 1 and 2. Several of the current SRs also cannot be performed in Modes 3 and 4. The footnote to SR 3.8.4.8 would also be deleted. There would also be renumbering in several of the SR notes.

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

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

The emergency diesel generators (DGs) and their associated emergency loads are accident mitigating features, rather than accident initiating equipment. Each DG is dedicated to a specific vital bus and these buses and DGs are independent of each other. There is no common mode failure provided by the testing changes proposed in this license amendment request (LAR) that would cause multiple bus failures. Therefore, there will be no significant impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. The changes include an increase in the online time the DG will be paralleled to the grid in Mode 1, 2, 3, [and] 4. The overall time that the DG is paralleled in all modes (outages/non-outage) should remain unchanged. As such, the ability of the DGs to respond to a design basis accident (DBA) can be adversely impacted by [the] proposed changes. However, the impacts are not considered significant based on the DG under test maintaining its ability to respond to an auto-start signal were one to be received during testing, along with the ability of the remaining DG to mitigate a DBA or provide a safe shutdown, and data that shows that the DG itself will not perturb the electrical system significantly. Furthermore, the proposed amendments for surveillance requirements (SR) 3.8.1.10 and SR 3.8.1.14 share the same electrical configuration alignment to the current monthly 1-hour loaded surveillance.

For SR 3.8.1.13, the DG would still be able to respond to an auto-start signal were one to be received during testing. The unavailability of the DG during the conduct of this SR 3.8.1.13 is minimal (approx[imately] 30 minutes) and is considered insignificant from a risk perspective.

In addition, operating experience and evaluation of the probability of a DG being rendered inoperable concurrent with or due to a significant grid disturbance, support the conclusion that the proposed changes in this LAR do not involve any significant increase in the likelihood of a safety-related bus blackout.

SR changes that are consistent with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF–283, Revision 3 and NUREG–1432, Revision 2 have been approved by the NRC, and the on-line tests allowed by the TSTF and the NUREG are only to be performed for the purpose of establishing operability [of the DG being tested]. Performance of these SRs during previously restricted modes will require an assessment to assure plant safety is maintained or enhanced.

The deletion of the footnote associated with SR 3.8.4.8 is an editorial change. This footnote was associated with coming out of the ninth refueling outage for Unit 1, which has since passed.

Therefore, the proposed change[s 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[s] would create no new accidents since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration currently allowed by other DG SRs that allow testing in plant Modes 1, 2, 3, and 4. This license amendment request does not impact any plant systems that are accident initiators or adversely impact any accident mitigating systems.

Therefore, the proposed change[s do] not create the possibility of a new or different accident from any accident previously evaluated.

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

The proposed changes do not involve a significant reduction in the margin of safety. The margin of safety is related to the ability of the fission product barriers to perform their design [safety] functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the plant DGs do not affect the operability requirements for the DGs, as verification of such operability will continue to be performed as required (except during different allowed modes [of operation]). Continued verification of operability supports the capability of the DGs to perform their required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Only one DG is to be tested at a time and the remaining DG will be available to safely [shut down] the plant or respond to a DBA, if required. Consequently, the performance of these fission product barriers will not be impacted by implementation of [the] proposed amendment.

In addition, the proposed changes involve no changes to [safety] setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted.

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

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

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: May 28, 2003.

Description of amendments request: The proposed amendment would revise the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, cooldown curves (Technical Specification Figure 3.4.3–2) to change the range of temperatures for which a cooldown rate of 100 °F/hr is acceptable.

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

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

In accordance with 10 CFR Part 50, Appendix G, the Calvert Cliffs pressure/temperature (P–T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials were developed using the methods of linear elastic fracture mechanics and the guidance found in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G. The proposed cooldown rates for the Technical Specification P–T limits were made possible by ASME Code Case N–640 which permits use of K_{IC} for reference stress intensity factor. [Temperatures that enable the low temperature overpressure protection system are not affected].

The proposed change only changes the temperature at which the cooldown transitions from 100°F/hr to 40°F/hr. It does not change the basic cooldown rates or methods of cooling down the Reactor Coolant System. This cooldown transition does not affect the probability of an accident previously evaluated because the cooldown rates have not changed. Additionally, since the cooldown rates are not changed above 300°F, the safety analyses and dose consequences in the Updated Final Safety Analysis Report are not affected.

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

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

The implementation of the proposed revision has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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

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

The margin of safety is defined by compliance with 10 CFR Part 50, Appendix G, requirements for adequate margin to prevent brittle failure of the reactor coolant pressure boundary materials. As discussed above, use of K_{IC} with continuous cooldown results in a conservative cooldown rate that will maintain plant safety. With the proposed change, the underlying intent of the 10 CFR Part 50, Appendix G, is maintained.

Therefore, this proposed change does not significantly reduce a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: May 28, 2003.

Description of amendment request: The proposed amendment would modify the Technical Specifications requirements for spent fuel storage pool boron concentration and fuel storage. The proposed amendment would eliminate the need to credit Boraflex neutron absorbing material for reactivity control in the H. B. Robinson Steam Electric Plant, Unit No. 2, spent fuel storage pool. The new analyses submitted by the licensee take credit for a combination of soluble boron and controlled fuel loading patterns within the spent fuel storage pool in order to maintain acceptable margins of subcriticality.

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

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

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

The proposed changes do not modify the facility. They apply additional administrative controls for maintaining the required boron concentration in the spent fuel storage pool. They also revise the acceptance criteria for the spent fuel storage pool criticality analyses. There will be a procedural change requiring increased frequency of spent fuel storage pool sampling for boron analysis. The sampling is performed in accordance with approved procedures and does not impact the probability or consequences of spent fuel storage pool accidents, which are a fuel handling accident and a loss of spent fuel storage pool cooling. The changes will allow for the further degradation of the Boraflex within the high density racks. The existence or degradation of the Boraflex has no relationship to the probability or consequences of a fuel handling accident or a loss of spent fuel storage pool cooling.

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 Previously Evaluated.

The proposed changes are related to the possibility of a criticality accident in the spent fuel storage pool. Detailed analyses have been performed to ensure a criticality accident in the spent fuel storage pool is not a credible event. The events that could lead to a criticality accident are not new. These events include a fuel mis-positioning event, a fuel drop event, and a boron dilution event. The proposed changes do not impact the probability of any of these events. The detailed criticality analyses performed demonstrate that criticality would not occur following any of these events. For the more likely events, such as a fuel mis-positioning event, k_{eff} remains less than or equal to 0.95. For the unlikely event that the spent fuel storage pool boron concentration was reduced to zero, k_{eff} remains less than 1.0. Since a criticality accident remains ''not credible,'' the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes continue to provide the controls necessary to ensure a criticality event could not occur in the spent fuel storage pool. The acceptance criteria are consistent with the acceptance criteria specified in 10 CFR 50.68, which provide an acceptable margin of safety in regard to the potential for a criticality event. Therefore, the changes do not result in a significant reduction in the margin of safety.

Based on the above discussion, [Carolina Power & Light Company] has determined that the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: March 4, 2003, as supplemented May 13, 2003.

Description of amendment request: The proposed amendment would revise selected sections of the Technical Specifications (TSs) based upon a re-analysis of fuel handling accidents (FHAs). The revised analysis is based upon selective implementation of the alternative source term (AST) methodology of Regulatory Guide (RG) 1.183, and in accordance with Title 10 of the Code of Federal Regulations (10 CFR), Section 50.67. Specifically, the amendment would revise: TS 3.7.8, ''Plant Systems, Control Room Envelope Pressurization System;'' TS 3.9.4, ''Refueling Operations, Containment Building Penetrations;'' TS 3.9.9, ''Refueling Operations, Containment Purge and Exhaust Isolation System,'' and TS 3.9.12, ''Refueling Operations, Fuel Building Exhaust Filter System.''

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 staff's review is presented below:

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

The proposed changes do not involve physical modifications to the plant equipment and do not change the operational methods or procedures used for the physical movement of fuel in containment or in the fuel building. As such, the proposed changes have no effect on the probability of occurrence of any accident previously evaluated.

The proposed changes are based upon the re-analysis of an FHA in the containment and an FHA in the fuel building area. The consequences of the re-analyzed events are expressed in terms of total effective dose equivalent (TEDE), and are not directly comparable to either the thyroid or whole body doses reported in the existing analyses. However, even taking this comparison into consideration, any dose increase is considered not to be significant as the revised analyses results meet the applicable TEDE acceptance criteria for AST implementation.

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

The containment closure components (*e.g.*, equipment access hatch, personnel access hatch doors, and various containment penetrations) and filtration systems are not accident initiators. The proposed changes do not involve the addition of new systems or components nor do they involve the modification of existing plant systems. The proposed changes do not change the operational modes or procedure used for the physical movements of fuel in containment or in the fuel building. The proposed changes do not affect the way in which an FHA is postulated to occur. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The margin of safety for the dose consequence analysis is considered to be that provided by meeting the applicable regulatory limits. The dose consequences of the existing FHA are within regulatory limits for whole body and thyroid doses as established in 10 CFR 100. The revised FHA using the AST method demonstrates that the dose consequences are within the regulatory limits for TEDE established in 10 CFR 50.67 and RG 1.183. There is no direct correlation between the old margins of safety established by meeting 10 CFR Part 100 and those established by the proposed change. The staff concludes, however, that meeting 10 CFR 50.67 and RG 1.183 limits would result in doses that would be within the 10 CFR Part 100 limits. Therefore, it is concluded that a reducation in margin of safety, if any, would not be significant.

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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: February 25, 2003, as supplemented June 9, 2003.

Description of amendment request: The proposed amendments require a Steam Generator (SG) Program that defines a performance based approach to maintaining SG tube integrity. The SG Program includes performance criteria that define the basis for tube integrity and provides reasonable assurance that SG tubing will remain capable of fulfilling its safety function of maintaining reactor coolant pressure boundary (RCPB) integrity. The proposed amendments add a new Technical Specification (TS) for SG Tube Integrity (3.4.18) and revise the TSs for Reactor Coolant System (RCS) Operational Leakage (3.4.13), SG Tube Surveillance Program (5.5.9), and SG Tube Inspection Report (5.6.8).

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:

Operation of the facility in accordance with the proposed amendments:

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

The proposed changes require a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents.

The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

The structural integrity performance criterion is a new requirement. It is included in the proposed SG Program administrative TS 5.5.9.

The accident induced leakage criterion is a new requirement. It is included in the proposed SG Program administrative TS 5.5.9.

The operational leakage criterion is equivalent to the existing requirement. Its limit is part of the proposed RCS Operational Leakage TS 3.4.13.

A SG tube rupture event is one of the design basis accidents analyzed as part of Catawba's licensing basis. In the analysis of a SG tube rupture event, a bounding primary to secondary leakage rate equal to the operational leakage rate limit in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube is assumed. For other design basis accidents, the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These analyses assume that primary to secondary leakage through each SG is 150 gallons per day.

The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis. The SG performance criteria proposed as part of these TS amendments identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed changes to TS 5.5.9. The program, defined by NEI [Nuclear Energy Institute] 97–06, ''Steam Generator Program Guidelines,'' includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

Probability of an Accident

The TS proposed by these license amendments define the actions required upon failure to maintain SG tube integrity and the surveillances necessary to verify that tube integrity is maintained. The proposed administrative TS contain performance criteria, repair criteria, repair methods, maximum SG inspection intervals, and reporting requirements. The set of TS proposed is a significant improvement over the existing SG TS.

In addition, the SG Program required by these amendments includes provisions important in satisfying the TS requirements. The topics addressed by the SG Program include:

- SG performance criteria, including an operational leakage limit,
- SG repair criteria and repair methods,
- SG inspection intervals, and
- Performance based SG inspections that include pre-inspection degradation assessments, condition monitoring assessments, operational assessments, and non-destructive examination technique requirements.

These SG Program provisions establish requirements that are an improvement as compared to the requirements in the existing TS. As an example, the SG Program requires an operational assessment that defines the maximum SG inspection interval that provides reasonable assurance that the performance criteria will continue to be met at the next inspection. The actual inspection interval is always chosen to be less than the interval determined by the operational assessment. The existing TS have no similar requirement. As a result, the function and integrity of the tubes are maintained with greater assurance and the probability of a SG tube rupture is decreased.

Consequences of an Accident

The consequences of design basis accidents are, in part, functions of the dose equivalent I^{131} in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the plant TS for operational leakage and for dose equivalent I^{131} in primary coolant to ensure the plant is operated within its analyzed condition.

The analysis of the associated design basis accidents assumes that the initial primary to secondary leak rate is 150 gallons per day in each SG (except for the ruptured SG in a SG tube rupture), and that the reactor coolant activity levels of dose equivalent I^{131} are at the TS values before the accident. The TS limits, license conditions, and other controls on I^{131} are unchanged by these amendment requests. These other controls include License Amendments 159 and 151 for Catawba Units 1 and 2, respectively, and the Catawba license amendment request submittal dated May 9, 2002, which is presently being reviewed by the NRC.

In addition, the proposed amendments include a new performance criterion for accident induced leakage that requires that the primary to secondary leakage resulting from an accident other than a SG tube rupture not exceed the value assumed in the dose analyses (150 gallons per day through each SG).

Since the proposed operational leakage limit is equivalent to the existing value, and since the proposed amendments include a new performance criterion for accident induced leakage, the proposed amendments will not increase the consequences of an accident.

From the above discussion, it is concluded that the proposed amendments do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the existing TS and enhances the requirements for SG inspections. The proposed TS changes do not adversely impact any other previously evaluated design basis accident and represent an improvement over the existing TS. Therefore, the proposed changes do not affect the consequences of a SG tube rupture accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of other accidents.

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

The proposed performance based requirements are an improvement over the requirements imposed by the existing TS. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed amendments do not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed changes do not impact any other plant system or component. The changes enhance SG inspection requirements. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The SG tubes in pressurized water reactors are an integral part of the RCPB and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

SG tube integrity is a function of the design, environment, and physical condition of the tube. The proposed license amendments do not affect tube design or operating environment. The proposed changes are expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the existing TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed revisions to the TS.

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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: May 21, 2003.

Description of amendment request: Change the technical specifications by extending the functional test frequency of the reactor protection system (RPS) intermediate range monitor (IRM) functions from weekly to 31 days, and to add more restrictive requirements for the RPS IRM—High Flux function.

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

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

The proposed changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. The change does not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the safety analysis. The changes do not impact the way in which surveillances are performed or introduce any accident initiators. The availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident are not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, or the high reliability of the equipment. More stringent requirements that ensure operability of equipment do not affect the initiation of any event, nor do they negatively impact the mitigation of any event.

Therefore, the proposed changes do 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?

The proposed changes do not introduce any failure mechanisms of a different type than previously evaluated, since no physical changes to the plant are being made. No new failure modes are introduced as no new or different equipment is being installed, and no installed equipment is being operated or surveillance tested in a different manner.

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

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

Although the proposed changes would result in changes to the interval between certain surveillance tests, the impact, if any, on system availability is minimal, based upon other more frequent testing that is performed, the existence of redundant systems and equipment, or overall system reliability. The changes do not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The imposition of more stringent requirements has no negative impact on margins of safety.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 30, 2003.

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

The changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated May 30, 2003.

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

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

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

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: Mr. Jonathan Rogoff, General Counsel, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: June 3, 2003.

Description of amendment request: The proposed amendment would revise the Palisades Plant Operating License and Technical Specifications to increase the licensed rated power level by 1.4 percent from 2530 megawatts thermal (MWt) to 2565.4 MWt. This power level increase is considered a measurement uncertainty recapture power uprate.

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

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

The proposed increase in power level is achieved by the taking credit for the accuracy of the existing feedwater flow measurement instrumentation, including the Crossflow ultrasonic flow measurement (UFM) system, which results in a more accurate feedwater flow used in the heat balance calculation. The increased flow accuracy utilizing the Crossflow UFM system improves the uncertainty in the core power level from the existing 2 percent margin to $\leq 0.5925\%$. The probability of an accident previously evaluated is not increased by the proposed change because the flow measurement instrumentation is not an initiator of design-basis accidents evaluated in the updated final safety analysis report [FSAR].

The plant design and licensing basis has been evaluated for operation at the proposed increased value of 2565.4 Megawatts thermal (MWt). All systems and components continue to acceptably perform their structural and operational functions.

There are no changes as a result of the proposed measurement uncertainty recapture power uprate to the design or operation of the plant that could affect system, component, or accident mitigative functions. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits.

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

The reduction in power measurement uncertainty is bounded by the safety analyses since they were performed or evaluated at 2580.6 MWt. Radiological consequences of [FSAR] Chapter 14 accidents were assessed previously and continue to be bounding. The FSAR Chapter 14 analyses continue to demonstrate compliance with the relevant accident analysis acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the proposed uprated power level. The proposed change has no adverse effects on any safety-related systems or component and does not challenge the performance or integrity of any safety-related system. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The maximum steady-state reactor power of 2580.6 MWt assumed in the accident analysis, including uncertainties, remains the same as previously analyzed. Therefore, the change in rated thermal power to 2565.4 MWt does not involve a significant reduction in the margin of safety.

The current accident analyses and system and component analyses had been previously performed at core powers that exceed the proposed measurement uncertainty recapture uprated core power. Evaluations have been performed for analyses that were done at nominal core power and have been found acceptable for the proposed measurement uncertainty recapture power uprate. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been either reviewed and approved by the Nuclear Regulatory Commission or are in compliance with applicable regulatory review guidance and standards. The proposed variable high power trip allowable value will ensure that the maximum actual steady state power at which a trip would be actuated is within safety analysis limits. 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: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: May 29, 2003.

Description of amendment requests: The License Amendment Request (LAR) revises TS 3.8.1, ''AC Sources—Operating'' to allow surveillance testing of the onsite standby emergency diesel generators (DG) during modes in which it is currently prohibited. Specifically, the licensee proposes removing the mode restrictions for the following surveillance requirements (SRs): SR 3.8.1.10 (full load rejection test), SR 3.8.1.13 (protective-trip bypass test), and SR 3.8.1.14 (endurance and margin test). This LAR also incorporates changes included in the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF–283, Revision 3. These changes modify the Notes in SRs 3.8.1.8 (transfer of AC sources test), 3.8.1.9 (post accident load rejection test), 3.8.1.11 (simulated loss of offsite power test), 3.8.1.12 (auto-start on safety injection (SI) signal test), 3.8.1.16 (restoration of loads to offsite power test), 3.8.1.17 (verification of test mode override test), 3.8.1.18 (engineered safety feature and auto-transfer load sequencing test), 3.8.1.19 (loss of offsite power plus SI signal response test), 3.8.4.7 (battery service test), and 3.8.4.8 (battery discharge test) to allow performance of the surveillances in order to reestablish operability following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated operability concerns during plant operation.

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

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

The emergency diesel generators (DGs) and their associated emergency loads are accident-mitigating features. As such, testing of the DGs themselves is not associated with any potential accident initiating mechanism. Each DG is dedicated to a specific vital bus and these buses and DGs are independent of each other. There is no common mode failure provided by the testing changes proposed in this license amendment request (LAR) that would cause multiple bus failures. Therefore, there will be no significant impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes.

The changes include an increase in the online time the DG will be paralleled to the grid in Mode 1 or 2. However, the overall time that the DG is paralleled in all modes (outage/non-outage) should remain unchanged. As such, the ability of the DGs to respond to a design basis accident can be adversely impacted by these proposed changes. However, the impacts are not considered significant based on the ability of the remaining two DGs to mitigate a design bases accident (DBA) or provide a safe shutdown, and data that shows that the DG itself will not perturb the electrical system. Furthermore, the proposed amendments for surveillance requirement (SR) 3.8.1.10 and SR 3.8.1.14 share the same electrical configuration alignment to the current monthly 1-hour loaded surveillance.

For SR 3.8.1.13, the DG would still be able to respond to an auto-start signal were one to be received during testing. The unavailability of the DG during the conduct of this SR 3.8.1.13 is minimal (approximately 5 minutes) and is insignificant from a risk perspective.

In addition, operating experience and evaluation of the probability of a DG being rendered inoperable concurrent with or due to a significant grid disturbance support the conclusion that the proposed changes in this LAR do not involve any significant increase in the likelihood of a safety-related bus blackout.

SR changes that are consistent with Industry/Technical Specification Task Force (TSTF) Standard Technical Specification (STS) change TSTF–283, Revision 3 have been approved by the NRC and the online tests allowed by the TSTF are only to be performed for the purpose of establishing operability. Performance of these SRs during normally restricted modes will require an assessment to assure plant safety is maintained or enhanced.

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

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

The proposed change would create no new accidents since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration currently allowed by other DG SRs that allow testing in plant Modes 1 and 2 and 3. This license amendment request does not impact any plant systems that are accident initiators or adversely impact any accident mitigating systems.

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

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

The proposed change does not involve a significant reduction in the margin of safety. The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the plant DGs do not affect the operability requirements for the DGs, as verification of such operability will continue to be performed as required (except during different allowed modes). Continued verification of operability supports the capability of the DGs to perform their required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 5, 2003.

Brief description of amendments: The proposed change involves the extension from 1 hour to 24 hours of the completion time (CT) for Condition B of Technical Specification (TS) 3.5.1, which defines requirements for accumulators. Accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. Condition B of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the Technical Specification Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved topical report WCAP–15049–A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated June 5, 2003.

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

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

The basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049–A is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049–A for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174, ''An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' and 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049–A, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049–A evaluation as a worst case data point. In addition, WCAP–15049–A states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049–A evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049–A evaluation demonstrates that the small increase in risk due to increasing the CT for an inoperable accumulator is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

Therefore, 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 proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049–A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049–A evaluation.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: June 11, 2003.

Description of amendment requests: The license amendment request proposes to revise Technical Specification (TS) 3.1.7, ''Rod Position Indication,'' TS 3.2.1, ''Heat Flux Hot Channel Factor,'' TS 3.2.4, ''Quadrant Power Tilt Ratio,'' and TS 3.3.1, ''Reactor Trip System Instrumentation,'' to allow use of a power distribution monitoring system as described in WCAP–12472–P–A, ''BEACON Core Monitoring and Operations Support System,'' for power distribution measurements.

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 power distribution monitoring system (PDMS) performs continuous core power distribution monitoring. This system utilizes the NRC-approved Westinghouse proprietary computer code, the Best Estimate Analyzer for Core Operations—Nuclear (BEACON), to provide data reduction for incore flux maps, core parameter analysis, load follow operation simulation, and core prediction. It in no way provides any protection or control system function. Fission product barriers are not impacted by these proposed changes. The proposed changes occurring with PDMS will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident. The changes associated with the PDMS do not affect plant systems such that their function in the control of radiological consequences is adversely affected. These proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Final Safety Analysis Report Update (FSARU).

Continuous on-line monitoring through the use of PDMS provides significantly more information about the power distributions present in the core than is currently available. This results in more time (*i.e.*, earlier determination of an adverse condition developing) for operator action prior to having an adverse condition develop that could lead to an accident condition or to unfavorable initial conditions for an accident.

Each accident analysis addressed in the Diablo Canyon Power Plant FSARU is examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of reload cores are bounded by previously accepted analyses. This examination, which is performed in accordance with the requirements set forth in 10 CFR 50.59, ''Changes, tests and experiments,'' ensures that future reloads will not involve a significant increase in the probability or consequences of any accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequence 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 implementation of the PDMS has no influence or impact on plant operations or safety, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The possibility for a new or different type of accident from any accident previously evaluated is not created since the changes associated with implementation of the PDMS do not result in a change to the design basis of any plant component or system. The evaluation of the effects of using the PDMS to monitor core power distribution parameters shows that all design standards and applicable safety criteria limits are met.

The proposed changes do not result in any event previously deemed incredible being made credible. Implementation of the PDMS will not result in more adverse conditions and will not result in any increase in the challenges to safety systems. The cycle specific variables required by the PDMS are calculated using NRC-approved methods. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded.

The proposed change, therefore, 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.

The margin of safety is not affected by the implementation of the PDMS. The margin of safety provided by current TS remains unchanged. The proposed changes continue to require operation within the core limits that are based on NRC-approved reload design methodologies. Appropriate measures exist to control the values of these cycle-specific limits. The proposed changes continue to ensure that appropriate actions will be taken if limits are violated. These actions remain unchanged.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 17, 2003.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) Surveillance Requirement 4.6.2.1.b.2.b. This change would remove the requirement to verify that the reactor thermal power output is less than, or equal to, 1% of rated thermal power when the suppression chamber average water temperature is above 95 °F. Additionally, the amendment would correct two typographical errors on TS index page ''x.''

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Section

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

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

Response: No.

The proposed changes do not affect the allowable suppression chamber average water temperatures provided in the TS. The changes do not affect previously evaluated events described in the UFSAR [Updated Final Safety Analysis Report] including all DBAs [Design Basis Accidents] and other operational transients.

The surveillance is extraneous because Action b of LCO [Limiting Condition for Operation] 3.6.2 directs the plant operators to commence a plant shutdown if the suppression chamber temperature cannot be restored. These changes do not affect plant systems, structures or components (SSCs).

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

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

Response: No.

The proposed changes do not affect the design function or operation of a plant SSC. No physical or procedural changes are associated with this LCR [License Change Request]. As a result, no new credible failure mechanisms, malfunctions, or accident initiators are related to this change. Additionally, no new modes of plant operation are created.

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

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

Response: No.

The proposed changes include the deletion of a surveillance requirement. This change is prompted by an LCO action statement, which prevents the plant from performing the surveillance. As a result, this change does not impact safety margins specified in the Hope Creek licensing basis.

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

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

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

NRC Section Chief: James W. Clifford.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: May 21, 2003.

Description of amendment request: The proposed amendment would change the source term for the Dose Calculation Methodology to the Alternate Source Term (AST). This change would result in design modifications to the Control Room Emergency Air Treatment System (CREATS), eliminate the requirement for the Containment Post Accident Charcoal Filters, and revise both the reactor coolant dose equivalent I–131 specific activity limit and the containment spray NaOH concentration limit.

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

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

Response: No.

The function of the CREATS is to provide a safe environment for the operators in the event of an accident, and thereby allow them to perform their accident mitigation responsibilities. The physical changes to the CREATS were designed to enhance the ability of the system to perform that function. The new system is an improvement in reliability, redundancy and leak tightness over the existing system. The change in design has no impact on accident initiation frequencies. Therefore the physical changes to the plant do not increase the probability or consequences of a previously evaluated accident.

The proposed Technical Specification changes involving the CREATS reflect the new system configuration and current industry guidance. The specifications ensure system functionality and protection of the operators under postulated accident conditions.

The new dose analysis indicates that the radiation dose to the operators and the public is acceptable without crediting the post accident charcoal filters removed from Technical Specification 3.6.6 and 5.5.10, and also bounds the change to the Reactor Coolant System activity limits in Technical Specification 3.4.16. The change to the dose conversion factor definition in Technical Specification [S]ection 1.1 is consistent with the new analysis.

The reference to ICRP–30 [International Commission on Radiological Protection Publication No. 30] in the Dose Equivalent I–131 definition is consistent with the new analysis and Standard Tech Specs, NUREG1431, [''Standard Technical Specifications Westinghouse Plants.'']

All calculated doses are within the regulatory limits prescribed in 10CFR50.67. In addition, with the exception of one calculated Exclusion Area Boundary (EAB) dose, all dose numbers are within the guidelines of Reg Guide 1.183, [''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,''] and Standard Review Plan (SRP) 15.0.1. This above-mentioned dose is in one particular direction from the source. The associated accident is the Locked Rotor Accident, which was not previously evaluated for dose at Ginna. The 100% fuel failure assumption used in this accident is widely considered to be overly conservative. Additionally, extra margin is built into the calculation because RG&E [Rochester Gas & Electric Corporation] assumed 500 gallons per day (GPD) of Steam Generator (SG) tube leakage per SG. Since the primary release pathway for this accident is SG tube leakage, and Reg Guide 1.183 (reference 3) allows an assumed tube leakage equal to the Tech Spec allowable leakage (~150 GPD/SG at Ginna), RG&E assumed a release rate of ~3.3 times greater than required. The calculated dose (2.7 Rem) is well below the regulatory limit of 25 Rem and only slightly greater than the published guideline of 2.5 Rem. Given the localized nature, associated probability/risk, and conservatism in this analysis, the calculated dose is considered acceptable.

Iodine removal was not credited in the existing analysis of doses for Equipment Qualification. Therefore, even though the Containment Post Accident Charcoal Filters will be removed from Tech Specs as a result of this amendment, it is not necessary to re-analyze these doses.

The Toxic Gas in-leakage analysis is bounded by the assumed in-leakage in the dose analysis. The amendment also does not hinder or change the ability to mitigate smoke infiltration as described in NEI [Nuclear Energy Institute] 99–03, Control Room Habitability Guidance.

This change has no impact on accident initiators, will not affect the ability of the operators to perform their designated functions, and removal of the requirement for CNMT [Containment] Post Accident Charcoal Filters is shown to be acceptable. 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.

For the proposed changes, a different kind of accident would involve a situation where the operators would become incapacitated or otherwise be prevented from fulfilling their function. The new system differs in that the cooling in the emergency mode is from direct expansion of R–22 refrigerant. A rupture of the coils could introduce the refrigerant into the Control Room environment. However, the charge of refrigerant R–22 in cooling system will be limited such that a rupture in the cooling coils would not exceed nationally accepted toxicity standards.

The radiation and/or toxic gas exposures are shown to be acceptable, and the ability of the plant to mitigate smoke infiltration has not changed. The new system will improve the environmental conditions in most situations and actually enhance the ability of the operators to perform their functions.

Given the above, an event that would result in preventing the operators from fulfilling their safety functions is not introduced by this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The new analysis was performed without crediting the existing Containment Post Accident Charcoal Filters and indicated that the Control Room and off-site doses remain within the required limits. Removal of the Post Accident Charcoal Filters from Technical Specification will not impact the operators' ability to function or significantly increase dose to the public.

The new Technical Specification surveillance limits for NaOH tank level and concentration establish criteria acceptable to meet the assumptions in the dose analysis.

The changes to the VFTP [Ventilation Filter Testing Program] program in Technical Specification reflect the removal of the Containment Post Accident Charcoal Filters consistent with the analysis, and the surveillance limits consistent with the new CREATS design.

The use of AST represents a change to a standardized and accepted dose calculation method.

The function of the CREATS system is to protect the operators and allow them to perform the necessary accident mitigation tasks. The proposed changes to the CREATS enhance this ability through improved redundancy and system operation. The analysis demonstrates that the Control Room will remain within prescribed limits during the design basis accidents. The operators will be able to perform their function and the public will be protected.

Therefore, the proposed change does not involve a significant reduction in a margin to 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: May 22, 2003 (TSC 03–02).

Description of amendment request: The proposed amendment would revise the limiting condition for operation for Technical Specification (TS) Section 3.7.5, ''Ultimate Heat Sink.'' This revision would modify the required minimum ultimate heat sink (UHS) water elevation in TS 3.7.5.a from 670 feet to 674 feet. The maximum emergency raw cooling water (ERCW) temperature requirement in TS 3.7.5.b will be increased from 83 degrees Fahrenheit (°F) to 87 °F. Limiting condition for operation requirements that are now obsolete because of the proposed changes are being deleted, as well as expired footnote provisions.

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

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

No. The proposed change to increase the UHS maximum temperature and the minimum water level does not alter the function, design, or operating practices for plant systems or components. The UHS is utilized to remove heat loads from plant systems during normal and accident conditions. This function is not expected or postulated to result in the generation of any accident and continues to adequately satisfy the associated safety functions with the proposed changes. Therefore, the probability of an accident presently evaluated in the safety analyses will not be increased because the UHS function does not have the potential to be the source of an accident and no plant equipment is altered as a result of this change. The heat loads that the UHS is designed to accommodate have been evaluated for functionality with the higher temperature and elevation requirements. The result of these evaluations is that there are existing margins associated with the systems that utilize the UHS for normal and accident conditions. These margins are sufficient to accommodate the postulated normal and accident heat loads with the proposed changes to the UHS. Since the safety functions of the UHS are maintained, the systems that ensure acceptable offsite dose consequences will continue to operate as designed. Therefore, the proposed changes to TS 3.7.5 will not significantly increase the consequences of an accident previously evaluated based on safety functions continuing to meet their accident mitigation requirements and limiting dose consequences to acceptable levels.

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

No. The UHS function is not an initiator of any accident and only serves as a heat sink for normal and upset plant conditions. By allowing the proposed change in the UHS temperature and elevation requirements, only the parameters for UHS operation are changed while the safety functions of the UHS and systems that transfer the heat sink capability continue to be maintained. The UHS function provides accident mitigation capabilities and does not reflect the potential for accident generation. Therefore, the possibility for creating a new or different kind of accident is not created because the UHS is only utilized for heat removal functions that are not a potential source for accident generation.

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

No. The proposed change has been evaluated for systems that are needed to support accident mitigation functions as well as normal operational evolutions. Operational margins were found to exist in the systems that utilize the UHS capabilities such that these proposed changes will not result in the loss of any safety function necessary for normal or accident conditions. The ERCW system has excess flow margins that will accommodate the increased flows necessary for the proposed temperature increase. While operating margins have been reduced by the proposed changes, safety margins have been maintained as assumed in the accident analyses for postulated events. Additionally, the proposed changes do not require the modification of component setpoints or operating provisions that are necessary to maintain margins of safety established by the SQN design. Therefore, a significant reduction in the margin to safety is not created by this proposed change.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: June 5, 2003 (TSC 03–07).

Description of amendment request: The proposed amendment would revise Action b of Technical Specification (TS) 3.6.1.9, ''Containment Ventilation System'' to allow an alternative to returning the inoperable containment purge supply or exhaust valve to operable conditions for continued operation. The alternative ensures isolation of the affected flow path such that potential release paths to the environment are sufficiently restricted to meet regulatory limits. This change will minimize the need to initiate a unit shutdown or delay start-up when acceptable means are available to ensure the required safety function.

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

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

No. The proposed change does not alter any plant system or operating practice. This change will allow the isolation of the affected flow path such that the safety function is completed when the associated automatic isolation valve is inoperable because of leakage. The containment purge supply and exhaust valves are not considered to be the source of an accident as their function is to isolate containment from the outside environs in the event of an accident. Accident generation probability is not affected by providing alternative isolation methods that continue to satisfy the required safety function. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

The proposed addition for the isolation of the affect flow path in place of a required shutdown of the unit, provides an equivalent safety function without the risk associated with a unit shutdown. Using a feature that has minimal potential for inadvertent loss of function and a more frequent surveillance to ensure that the isolation function is maintained, is as good or better than the automatic system that is required by the TSs. This is because the proposed action utilizes a passive feature in place of an active system and ensures offsite dose consequences within required limits. Additionally, the overall plant safety is enhanced by not requiring a unit shutdown when acceptable measures can be taken to preserve the safety function of the containment purge supply and exhaust valves. Therefore, the proposed change does not involve an increase in the consequences of an accident previously evaluated.

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

No. The proposed change does not involve a change to plant systems, components, or operating practices that could result in a change in accident generation potential. In addition, the purge and exhaust valves are utilized for the isolation of flow paths to the environs and are not a feature that could generate a postulated accident. Use of the proposed action for inoperable purge and exhaust valves will not impact the potential for accidents. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed changes do not alter plant systems or their setpoints that are used to maintain the margin of safety. Additionally, the proposed change provides a method to ensure the safety function of the containment ventilation and isolation systems are retained for accident mitigation purposes. The proposed change will improve the margin of safety by not requiring a unit shutdown when acceptable methods for maintaining plant safety functions can be achieved. Therefore, the proposed change does not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: May 30, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications to replace the single boron concentration requirement with a table that defines the minimum and maximum amount of boron that is required for accident mitigation based on the number of tritium producing burnable absorber rods (TPBARs) in the core.

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

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

Response: No.

The proposed change modifies the required boron concentration for the cold leg accumulators (CLAs) and RWST [Refueling Water Storage Tank]. The proposed values have been verified to maintain the required accident mitigation safety function for the CLAs and RWST. The CLAs and RWST safety function is to mitigate accidents that require the injection of borated water to cool the core and to control reactivity. These functions are not potential sources for accident generation and the modification of the boron concentration that supports event mitigation will not increase the potential for an accident. Therefore, the possibility of an accident is not increased by the proposed changes. The boron levels for this change are based on the number of TPBARs in the core. As the number of rods is increased the need for additional shutdown boron also increases. This effect has been evaluated with a similar methodology utilized for previously NRC approved amendments associated with tritium production. This methodology ensures that the impact of TPBARs is adequately compensated for by the required boron concentrations and has been incorporated into the proposed revision. Since the boron levels will continue to maintain the safety function of the CLAs and RWST in the same manner as currently approved, the consequences of an accident are not increased by the proposed changes.

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

Response: No.

The proposed change only modifies boron concentrations for accident mitigation functions of the CLAs and RWST. These functions do not have a potential to generate accidents as they only serve to perform mitigation functions associated with an accident. The proposed requirements will maintain the mitigation function in an identical manner as currently approved. There are no plant equipment or operational changes associated with the proposed revision other than the adjustment of the boron level in the CLAs and RWST. Therefore, since the CLA and RWST functions are not altered and the plant will continue to operate without change, the possibility of a new or different kind of an accident is not created.

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

Response: No.

This change proposes boron concentration requirements that support the accident mitigation functions of the CLAs and RWST equivalent to the currently approved limits. The proposed change does not alter any plant equipment or components and does not alter any setpoints utilized for the actuation of accident mitigation system or control functions. The proposed boron values have been verified to provide an adequate level of reactivity control for accident mitigation. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: June 5, 2003.

Brief description of amendments: The proposed change involves the extension from 1 hour to 24 hours of the completion time (CT) for Condition B of Technical Specification (TS) 3.5.1, which defines requirements for accumulators. Accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system if, as during a loss of coolant accident, the coolant pressure decreases to below the accumulator pressure. Condition B of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the Technical Specification Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved topical report WCAP–15049–A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated June 5, 2003.

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

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

The basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049–A is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049–A for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174, ''An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,'' and 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049–A, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049–A evaluation as a worst case data point. In addition, WCAP–15049–A states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049–A evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049–A evaluation demonstrates that the small increase in risk due to increasing the CT for an inoperable accumulator is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

Therefore, 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 proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049–A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049–A evaluation.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 6, 2003.

Brief description of amendments: The proposed license amendments would change Technical Specification (TS) Section 3.8.4, ''DC Sources—Operating,'' TS Section 3.8.5, ''DC Sources—Shutdown,'' and TS Section 3.8.6, ''Battery Cell Parameters,'' and add a new TS Section 5.5.19, ''Battery Monitoring and Maintenance Program'', to establish an administrative controls program for the maintenance and monitoring of the station safety-related batteries. The purpose of the proposed changes is to provide increased operational flexibility and allow more efficient application of plant resources to safety significant activities.

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

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

Response: No.

The proposed change affects TS sections 3.8.4 ''DC Sources—Operating,'' TS 3.8.5 ''DC Sources—Shutdown,'' TS 3.8.6 ''Battery Cell Parameters,'' and TS Administrative Controls section 5.5.

The proposed change restructures the TS for the DC electrical power subsystem and adds new Conditions and Required Actions with increased Completion Times to address battery charger inoperability. Neither the DC electrical power subsystem nor associated battery chargers are initiators of any accident sequence analyzed in the updated Final Safety Analysis Report (FSAR). Operation in accordance with the proposed TS ensures that the DC electrical power subsystem is capable of performing its function as described in the FSAR, therefore the mitigative functions supported by the DC electrical power subsystem will continue to provide the protection assumed by the analysis.

The relocation of preventive maintenance surveillance, and certain operating limits and actions to a newly-created, licensee-controlled TS [5.5.19], ''Battery Monitoring and Maintenance Program,'' will not challenge the ability of the DC electrical power subsystem to perform its design function. The maintenance and monitoring required by current TS, which are based on industry standards, will continue to be performed. In addition, the DC electrical power subsystem is within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure the control of maintenance activities associated with the DC electrical power subsystem.

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

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

Response: No.

The proposed change does not involve any physical alteration of the units. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints at which protective or mitigative actions are initiated that are affected by the proposed changes. The operability of the DC electrical power subsystem in accordance with the proposed TS is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. These proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures, which ensure the unit remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed changes do not alter assumptions made in the safety analyses.

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

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

Response: No.

The proposed change will not adversely affect operation of plant equipment and will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The changes associated with the new Battery Maintenance and Monitoring Program will ensure that the station batteries are maintained in a highly reliable manner. The equipment fed by the DC electrical system will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 9, 2003.

Description of amendment request: The proposed amendment would make administrative changes to Section 6 of the Surry Power Station Technical Specifications (TS) for Units 1 and 2 to adopt the format for topical report references that are described in Industry/Technical Specifications Task Force Traveller, TSTF–363, Rev 0, ''Revised Topical Report References in Improved Technical Specification (ITS) 5.6.5, COLR.''

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

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

The proposed change is administrative in nature and as such does not impact the condition or performance of any plant structure, system or component. The proposed administrative change does not affect the initiators of any previously analyzed event or the assumed mitigation of accident or transient events. As a result, the proposed change to the Surry Technical Specifications does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed administrative change.

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

The proposed change is administrative in nature, and therefore does not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions, and no new failure modes are being introduced. Therefore, the proposed administrative change to the Surry Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

The proposed change is administrative in nature and does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the proposed administrative change. Therefore, the proposed change to the Surry Technical Specifications does not involve any 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: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion

Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: November 16, 2001, as supplemented by letters dated October 4, 2002, and March 28, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.6.2.2, ''Suppression Pool Water Level,'' and TS 3.6.2.4, ''Suppression Pool Makeup System,'' to permit draining the reactor cavity pool portion of the upper containment pool in MODE 3, ''Hot Shutdown,'' with the reactor vessel pressure less than 235 psig.

Date of issuance: June 12, 2003.

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

Amendment No.: 156.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: April 2, 2002 (67 FR 15621). 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 amendment is contained in a Safety Evaluation dated June 12, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 26, 2002, as supplemented November 22, 2002.

Brief description of amendment: The amendments change the Technical Specifications for the pressure-temperature limits curves in Technical Specification 3.4.9, ''RCS Pressure and Temperature (P/T) Limits.''

Date of issuance: June 18, 2003.

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

Amendment No.: 228 and 256.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register***: August 6, 2002 (67 FR 50949). The November 22, 2002, supplement contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 18, 2003.

No significant hazards consideration comments received: No.

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

Date of application of amendments: October 24, 2002, as supplemented by letters dated November 21, 2002, and February 19, 2003.

Brief description of amendments: The amendments revise TS 3.5.3, Low Pressure Injection, Condition A, to change the Completion Time from 72 hours to 7 days. This revision will allow longer corrective maintenance to be completed at power, without requiring a plant shutdown. It will also reduce shutdowns due to a Limiting Condition for Operation requirement.

Date of Issuance: June 18, 2003.

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

Amendment Nos.: 332, 332, and 333.

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

*Date of initial notice in **Federal Register**:* December 24, 2002 (67 FR 78517).

The supplement dated November 21, 2002, did not change the scope of the October 24, 2002, application; however it did change the licensee's proposed No Significant Hazards Consideration Determination (NSHCD). The supplement dated February 19, 2003, provided clarifying information that did not change the scope of the October 24, 2002, application nor the initial proposed NSHCD.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 18, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 19, 2003.

Brief description of amendment: The amendment deletes Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminates the requirements to have and maintain the post accident sampling system at River Bend Station, Unit 1. The amendment also addresses related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: June 23, 2003.

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

Amendment No.: 134.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* April 29, 2003 (68 FR 22746).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 23, 2003.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Beaver County, Pennsylvania

Date of application for amendment: July 24, 2002, as supplemented February 4, 2003.

Brief description of amendment: The amendment changed the MSIV full-closure stroke time of Technical Specification (TS) surveillance requirement 4.7.1.5 from 5 seconds to 6 seconds. Additionally, the once-per-92-day requirement to part-stroke exercise the main steam isolation valves (MSIVs) was replaced with criteria to test each MSIV pursuant to TS 4.0.5, which requires testing in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI.

Date of issuance: June 25, 2003.

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

Amendment No: 137.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* September 17, 2002 (67 FR 58644). The supplement dated February 4, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 25, 2003.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: July 3, 2002, as supplemented September 24, 2002, January 10, 2003, and March 20, 2003.

Brief description of amendment: The amendment revises Improved Technical Specification (ITS) 3.8.1 and associated Bases, ''AC Sources-Operating,'' by extending the allowed outage time for the emergency diesel generators from 72 hours to 14 days.

Date of issuance: June 13, 2003.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance except for installation of an Aac source. An Aac source as described in the licensee's application supplement dated March 20, 2003, shall be installed before completion of refueling outage 14, as discussed in the NRC Safety Evaluation dated June 13, 2003. Implementation shall include incorporation of a description of the Aac source into the next scheduled Final Safety Analysis Report update after the Aac installation.

Amendment No.: 207.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* August 6, 2002 (67 FR 50955). The September 24, 2002, January 10, 2003, and March 20, 2003, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 19, 2003.

Brief description of amendment: The amendment deletes Technical Specification 6.8.C, ''Post Accident Sampling,'' and thereby eliminates the requirements to have and maintain the post accident sampling system at the Monticello Nuclear Generating Plant.

Date of Issuance: June 17, 2003.

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

Amendment No.: 136.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* May 13, 2003 (68 FR 25655).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 14, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 3/4.3.4.2 to extend the surveillance test intervals and allowed out-of-service times for the end-of-cycle recirculation pump trip system instrumentation. In addition, the TS Bases have been revised to address the proposed changes.

Date of issuance: June 24, 2003.

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

Amendment No.: 148.

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

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 24, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 26, 2002, as supplemented on March 20, 2003.

Brief description of amendments: The amendments revise setpoint and allowable values of the steam generator (SG) low-low level trip function in Technical Specification (TS) Table 2.2–1, ''Reactor Trip System Instrumentation Trip Setpoints,'' and TS Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation Trip Setpoints.'' The TS changes are necessary to account for a flow-induced pressure drop through the mid-deck plate inside the SG in the SG water level measurement.

Date of issuance: June 13, 2003.

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

Amendment Nos.: 257 and 238.

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

*Date of initial notice in **Federal Register***: February 4, 2003 (68 FR 5680). The March 20, 2003, supplement contained clarifying information and did not change the staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 2003.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket No. 50–364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendments request: March 31, 2003, as supplemented by letter dated April 29, 2003.

Brief description of amendments: The amendment modifies Technical Specifications (TS) Surveillance Requirement (SR) 3.4.11.1, for Farley, Unit 2 only by the addition of the following note that states, ''Not required to be performed for Unit 2 for the remainder of operating cycle 16 for Q2B31MOV8000B.'' In addition, a temporary TS SR 3.4.11.4 is added to provide compensatory action for this block valve while SR 3.4.11.1 is suspended. Further, this SR requires that power to the Farley, Unit 2 Power Operated Relief Valve Q2B31MOV8000B be checked at least every 24 hours for the remainder of Operating Cycle 16.

Date of issuance: June 13, 2003.

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

Amendment Nos.: 151.

Facility Operating License No. NPF–8: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register***: May 13, 2003 (68 FR 25658).

The supplement dated April 29, 2003, provided clarifying information that did not change the scope of the March 31, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 2003.

No significant hazards consideration comments received: No.

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

Date of amendments request: February 14, 2002, as supplemented by letters dated July 29, 2002 and March 27, 2003.

Brief description of amendments: The amendments revise STP technical specifications to eliminate shutdown actions associated with radiation monitoring instrumentation. The proposed changes will enhance plant reliability by reducing exposure to unnecessary shutdowns and increase operational flexibility, and relax certain other restrictions.

Date of issuance: June 9, 2003.

Effective date: As of the date of issuance to be implemented within 4 months from the date of issuance.

Amendment Nos.: Unit 1—153; Unit 2—141.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15629).

The July 29, 2002, and March 27, 2003, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** Notice (67 FR 15629) and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 9, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 4, 2002.

Brief description of amendments: The amendments revise several Limiting Conditions for Operation (LCO) Notes and Required Actions in the Technical Specifications that require suspension of operations involving positive reactivity additions or suspension of operations involving reactor coolant system boron concentration reductions. The amendments revise these LCO Notes and Required Actions to allow small, controlled, safe insertions of positive reactivity, but limit the introduction of positive reactivity such that compliance with the required shutdown margin or refueling boron concentration limits will still be satisfied.

Date of issuance: June 24, 2003.

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

Amendment Nos.: 105 and 105.

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

Date of initial notice in ***Federal Register***: January 7, 2003 (68 FR 813).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 24, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 30th day of June 2003.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 03–17028 Filed 7–7–03; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

July 17, 2003, Board of Directors Meeting; Sunshine Act

TIME AND DATE: Thursday, July 17, 2003, 1:30 p.m. (Open Portion). 1:45 p.m. (Closed Portion).

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

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

MATTERS TO BE CONSIDERED:

1. President's Report.
2. Testimonial D. Cameron Friday.
3. Approval of April 24, 2003 Minutes (Open Portion).

FURTHER MATTERS TO BE CONSIDERED:

(Closed to the Public 1:45 p.m.)

1. Finance Project in Brazil
2. Finance Project in Russia
3. Insurance Project in Croatia
4. Approval of April 24, 2003 Minutes (Closed Portion)
5. Pending Major Projects
6. Reports

FOR FURTHER INFORMATION CONTACT: Information on the meeting may be obtained from Connie M. Downs at (202) 336–8438.

Dated: July 3, 2003.

Connie M. Downs,

Corporate Secretary, Overseas Private Investment Corporation.

[FR Doc. 03–17344 Filed 7–3–03; 12:10 am]

BILLING CODE 3210–01–M

POSTAL SERVICE

Board of Governors; Sunshine Act Meeting

Board Votes to Close June 27, 2003, Meeting

By telephone vote on June 27, 2003, the Board of Governors of the United States Postal Service voted unanimously to close to public observation its meeting held in Washington, DC, via teleconference. The Board determined that prior public notice was not possible.

Items Considered

1. Personnel Matter.