Dated this 1st day of March 2000 at Rockville, Maryland.

For the Nuclear Regulatory Commission.

Ronald D. Hauber,

Deputy Director, Office of International Programs.

[FR Doc. 00–5586 Filed 3–7–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 12, 2000, through February 25, 2000. The last biweekly notice was published on February 23, 2000 (65 FR 9000).

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 Review and Directives Branch, Division of Freedom of Information and Publications 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 NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 7, 2000, 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 Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of amendment request: November 30, 1999.

Description of amendment request: The proposed amendment revises the test standard for laboratory testing of activated charcoal to tests in accordance with the ASTM D3803–1989 standard in response to Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

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:

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

The changes included in this request do not affect any accident initiating events. No new accident initiators or new failure modes are created. These changes will not result in any change to the charcoal efficiency credited in the accident analyses for any of the air treatment systems. The ability of each of the accident mitigation air treatment systems to perform its function will not be affected. System design flow requirements and filter/adsorber bank bypass leakage requirements remain unchanged. Therefore, the proposed changes will not adversely impact the capability of the accident mitigation air treatment systems and could not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

This LCA [license change application] does not involve the addition of any new hardware. The requested changes only affect testing standards for the three air treatment systems used for accident mitigation. Change[s] of a test standard for the air treatment systems could not create a new accident scenario. Therefore, these changes do not create the potential for any accident different from those that have been evaluated.

C. These proposed changes do not involve a significant reduction in a margin of safety.

The proposed T.S. changes will have no adverse affect on the performance of the three accident mitigation Air Treatment Systems. System design flow requirements and filter/adsorber bank bypass leakage requirements remain unchanged. Use of the charcoal lab testing protocol suggested by Generic Letter 99–02 will ensure that the charcoal adsorber is better able to adsorb radioiodine generated during postulated accidents. These changes do not result in a degradation of safety related equipment, and therefore, 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: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Marsha Gamberoni, Acting.

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: November 19, 1999.

Description of amendments request: The proposed amendments would revise Technical Specification (TS) 5.5.11.c, ''Ventilation Filter Testing Program (VFTP),'' to change the testing requirements of the engineered safety systems charcoal adsorbers.

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:

Standard 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 involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 5.5.11.c, initiates a laboratory performance test of adsorber carbon (charcoal) that yields more accurate results than what is currently required by TS. The proposed change also deletes the specific reference to the ANSI [American National Standards Institute] standard by which the adsorber carbon sample is obtained. The proposed changes to test adsorber carbon to a more current and improved ASTM [American Society for Testing and Materials] standard and delete the ANSI standard by which the adsorber carbon sample is obtained would not be plant accident initiators as described in Chapter 6 or Chapter 15 of the PVNGS [Palo Verde Nuclear Generating Station] UFSAR

[Updated Final Safety Analysis Report]. The changes would not involve a significant increase in the probability of an accident previously evaluated.

Carbon adsorption plays a direct role in mitigating the consequences of a radiological event. Safety-related air-cleaning units used in the ESF [engineering safety features] ventilation systems of nuclear power plants reduce the potential onsite and offsite consequences of a radiological accident by the adsorption of radioiodine. The proposed amendment to change the laboratory performance test for carbon will yield more conservative results than what is currently required by TS. Hence, it will better ensure that the adsorber carbon for TS systems used in the mitigation of an accident remains above the assumed carbon decontamination efficiency referenced in Chapter 6 and Chapter 15 of the UFSAR.

This proposed amendment does not alter, degrade, or prevent actions described or assumed in an accident. It will not alter any assumptions previously made in evaluating radiological consequences or, affect any fission product barriers. It does not increase any challenges to safety systems as well. Therefore, this proposed amendment would not significantly increase the consequences of an accident previously evaluated.

Standard 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 create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to TS 5.5.11.c, initiates a laboratory performance test of adsorber carbon that yields more accurate results than what is currently required by TS. The proposed changes to test adsorber carbon to a more current and improved ASTM standard and delete the specific reference to the ANSI standard by which the adsorber carbon sample is obtained would not be plant accident initiators as described in Chapter 6 or Chapter 15 of the PVNGS UFSAR. The proposed amendment does not change the function of any SSC [structure, system, and component]. TS nuclear air treatment systems function to filter radiological releases during design basis accidents. This change will provide greater assurance that this function is provided. The revised TS required laboratory tests utilize practices now in place, changing only the testing parameters. The changes do not alter, degrade, or prevent actions described or assumed in an accident described in the PVNGS UFSAR from being performed. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change does not involve a significant reduction in a margin of safety.

The margin of safety, as defined in the PVNGS Technical Specifications, is not reduced but is enhanced due to improved testing. This change initiates a laboratory performance test on adsorber carbon that yields more accurate results than what is currently required by TS and deletes the specific reference to the ANSI standard by which the adsorber carbon sample is obtained. The proposed change to test adsorber carbon to a more current and improved ASTM standard will ensure the carbon media's ability to adsorb radioactive gases will remain above that credited in the PVNGS' dose analysis for postulated accidents.

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: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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: December 1, 1999.

Description of amendments request: The proposed amendments to the operating licenses would delete or update outdated administrative information and delete license conditions that are no longer applicable or have been completed.

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:

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

No—The proposed administrative changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed administrative Operating License (OL) amendments would (1) delete or update operating license references to outdated administrative information, (2) delete license conditions that were complied with and are no longer applicable to the current operating environment, and (3) delete license conditions that were one-time requirements and have been completed. Since these proposed changes are administrative and have no [e]ffect on the current OL requirements, plant design, operation, or maintenance, the proposed administrative changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No—The proposed administrative changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes would have no [e]ffect on the physical plant. Consequently, plant configuration and the operational characteristics remain unchanged and the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No—The proposed administrative changes do not involve a significant reduction in a margin of safety. The proposed changes are administrative and have no [e]ffect on the current OL requirements, plant design, operation, or maintenance. No margin of safety would be affected by the proposed administrative changes to the PVNGS [Palo Verde Nuclear Generating Station] OLs since no current operating requirements would be changed.

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: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: January 27, 2000.

Description of amendments request: The proposed amendment modifies the conditions of containment closure during core alterations/fuel handling and loss of shutdown cooling in Calvert Cliffs Units 1 and 2. The reason for this proposed amendment is to improve personnel safety and the progress of outages by allowing greater egress from and access to the Containment during refueling outages. A new containment outage door assembly will be installed on the outside of the equipment hatch opening to provide quicker closure, improve safety when the door is open, and allow more flexibility when staging material in the Containment during an outage. Changes to the way the personnel air lock and the containment purge system are operated during maintenance activities on the Shutdown Cooling System are also part of the proposed amendment.

The proposed amendment changes Technical Specifications (TSs) 3.9.3 and 3.9.4 to allow the new containment outage door to remain open during core alterations and fuel handling, during maintenance and testing activities on the Shutdown Cooling system, and to be used as an alternate to the existing equipment hatch to close the equipment hatch opening when containment closure is required. The proposed changes will also allow the personnel air lock and the containment purge valves to remain open during maintenance activities on the Shutdown Cooling System. Baltimore Gas and Electric Company (BGE) also proposes to revise TS 3.9.3 to indicate that four bolts is the minimum number required to secure the equipment hatch for closure. In addition, BGE proposes deleting the words ''when there is 23 feet of water above the fuel'' from Limiting Condition for Operation 3.9.3.c.2 since this requirement is already part of the applicability statement.

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

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

The proposed changes will modify the conditions of containment closure during core alterations/fuel handling and during maintenance/testing activities on the Shutdown Cooling (SDC) System. Specifically, the proposed changes will allow the new containment outage door, the personnel air lock door, and the containment purge valves to stay open during core alterations/fuel handling, and during maintenance and testing activities on the SDC System. The proposed change will also allow the new containment outage door to be used as an alternate to the existing equipment hatch to close the equipment hatch opening when closure is required. Additionally, the proposed changes will change the wording of the Technical Specifications to indicate that four bolts is the minimum number required to secure the equipment hatch when it is used for containment closure. The proposed changes also remove[ ] the water level requirement from Limiting Condition for Operation 3.9.3 since the water level requirement is part of the applicability statement for this Technical Specification.

Closing the containment penetrations is considered to be a mitigator of the radiological consequences of a fuel handling incident and a loss of SDC, not an initiator. Therefore, allowing the containment outage door, personnel air lock, and the containment purge valves to be open during these outage activities does not involve a significant increase in the probability of an accident previously evaluated.

The consequence of a fuel handling incident is the release of radioactivity from Containment. The potential offsite dose resulting from a fuel handling incident has been evaluated. Based on a minimum decay time of 100 hours prior to handling fuel (Technical Reference Manual Section 15.9.1), the calculated offsite doses resulting from a fuel handling incident are 14.06 rem to the thyroid, and 0.457 rem to the whole body, with the personnel air lock door open. All activity released from Containment over the length of the incident is assumed to be unfiltered. The calculated doses resulting from a fuel handling incident are less than 25% of the limits of 10 CFR Part 100 (75 rem thyroid and 6 rem whole body). This analysis will apply to the equipment hatch opening because the analysis assumes no containment closure. The amount of radioactivity released is bounded by the current analysis of record. Although natural air circulation will cause some containment air to go out through any opening in a fuel handling accident, there is no pressure produced to push the radioactivity out of Containment. Therefore, having the containment outage door open during core alterations and fuel handling does not involve an increase in the consequences of an accident previously evaluated. Additionally, if the equipment hatch is to be used, specifying a minimum number of four bolts will allow the optional use of more bolts, if desired.

The consequence of a loss of SDC is the potential for release of radioactivity to the atmosphere outside Containment. Closing containment penetrations is a mitigator of that consequence. Administrative controls will be put in place to ensure that in an emergency containment closure can be quickly achieved. The emergency air lock will have at least one door closed when containment closure is required by a SDC condition. The containment purge system isolation valves are closed automatically on a containment high radiation signal and can be shut by remote manual operation. The maximum calculated pressure that can develop in the Containment for the limiting loss of SDC case is 12 psig. All required penetration closure devices can withstand that pressure. Therefore, allowing the personnel air lock doors, the containment outage door, and the purge isolation valves to remain open does not involve a significant increase in the consequences of a loss of SDC.

Therefore, the proposed Technical Specificaton changes do 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 type of accident from any accident previously evaluated.

This requirement change does not involve a significant change in the operation of the plant and no new accident initiation mechanism is created by the modification. Closing containment penetrations is considered to be a mitigator of the radiological consequences of any accident in the Containment, not an initiator. The equipment hatch opening, the personnel air lock, and the purge supply and exhaust are currently opened and closed during the course of an outage. The proposed changes allows them to remain open during a period when they are currently required to be closed. The closure function of the equipment hatch opening in Modes 5 and 6 will be performed by a hinged containment outage door; thus, closing the equipment hatch opening will be easier and will require fewer people and less time. The operation of the containment outage door is not a significantly different method of operation from that of other dogged doors at Calvert Cliffs. Using the containment outage door to close the equipment hatch opening instead of the equipment hatch also mitigates the consequences of the incident and does not initiate an accident.

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

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

The margin of safety for containment closure during core alternation/fuel handling is based on the amount of offsite dose resulting from a fuel handling incident and the safety of personnel in the Containment at the time of the incident. An offsite dose calculation previously approved by the NRC for a fuel handling incident is 14.06 rem to the thyroid, and 0.457 rem to the whole body, with no containment closure established, and any activity released from the Containment assumed to be unfiltered. These calculated doses are less than 25% of the limits of 10 CFR Part 100. The analysis will apply to the containment outage door because the analysis assumes no containment closure. Emergency personnel egress from Containment will be through the open door, which is an improvement in personnel safety because this exit is not currently available. Additionally, trained personnel will be available to close the door and contain any radiation released inside Containment as a result of a fuel handling incident. Leaving the containment outage door open during core alterations and fuel handling will not allow more than the calcutated amount of radionuclides to escape from Containment; shutting the door following a fuel handling incident will increase the margin of safety by keeping the actual offsite dose lower than the calculated dose.

Therefore, allowing the containment outage door to be open during fuel handling would not involve a significant reduction in the margin of safety.

The margin of safety for containment closure in the case of loss of SDC is twofold: (1) the time required to close the Containment to prevent a radioactive release to the atmosphere outside Containment if SDC should be lost; and (2) the ability to retain the pressure generated by boiling of reactor coolant as a result of a loss of SDC.

Currently, all containment penetrations are required to be closed prior to taking the SDC

System out-of-service for maintenance, or within four hours if SDC is lost. The radiological consequences of a loss of SDC incident do not occur immediately on loss of SDC. The containment purge isolation valves close rapidly on a high radiation signal or are closed by remote manual operation. The containment outage door and the personnel air lock doors are designed to be closed rapidly by site personnel. Other containment penetrations that could release radiation to the environment outside the Containment will be required to be closed. The maximum calculated pressure that can develop in the containment as a result of a loss of SDC is 12 psig. The purge isolation valves, the personnel air lock doors, and the containment outage door are all designed to meet this pressure retaining requirement. The proposed changes do not increase the possibility of a release of radiation following a loss of SDC incident.

Therefore, the ability to provide containment closure is maintained and the margin of safety is not significantly reduced by this proposed activity.

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: Marsha Gamberoni, Acting.

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

Date of amendment request: January 11, 2000.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected actuation instrumentation. The proposed amendments implement AOT/STI changes based on Topical Reports by General Electric Company and the Boiling Water Reactor Owners' Group that have previously been reviewed and approved by the NRC.

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:

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

The proposed TS changes increase the Allowable Outage Times and Surveillance Test Intervals (AOT/STI) for actuation instrumentation based on analyses developed and approved by the Nuclear Regulatory Commission (NRC). TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed changes do not involve any physical changes to plant systems, structures or components (SSCs), or the manner in which these SSCs are operated. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. As justified and approved in the AOT/STI licensing topical reports, the proposed changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analyses. The proposed changes establish or modify time limits allowable for operation with inoperable instrument channels based on analyses which have been approved by the NRC. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve any physical changes to SSCs, or the manner in which these SSCs function. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes increase the AOTs and STIs for actuation instrumentation based on generic analyses completed by the Boiling Water Reactors Owners' Group (BWROG). The NRC has reviewed and approved the generic studies and has concurred with the BWROG that the proposed changes do not significantly affect the probability of failure or availability of the affected instrumentation systems. The analysis determined that there is no significant change in the availability and/or reliability of instrumentation as a result of the proposed changes in AOTs and STIs.

Furthermore, the change to increase the frequency of the reactor protection system scram contactor testing has been shown to improve plant safety. ComEd has determined these studies are applicable to Dresden Nuclear Power Station, Units 2 and 3. The proposed changes to AOTs provide realistic times to complete required testing and maintenance actions without increasing the overall instrument failure frequency. Likewise, the extended STIs do not result in significant changes in the probability of instrument failure. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures. Therefore, it is concluded that the proposed changes will not result in a reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767

NRC Section Chief: Anthony J. Mendiola.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: November 22, 1999.

Description of amendment request: The proposed amendment would approve the use of new values for post-accident containment pressure in Pilgrim's net positive suction head (NPSH) analyses performed for the emergency core cooling system (ECCS) pumps.

Basis for proposed no significant hazards consideration determination:

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

(1) Will crediting the proposed post-LOCA [loss of coolant accident] containment pressure in ECCS analysis involve a significant increase in the probability or consequences of an accident previously evaluated?

Chapter 14 of the FSAR [final safety analysis report] contains evaluations of the worst postulated accidents that the Pilgrim plant was evaluated for, which include the refueling accident, the main steam line break outside primary containment, the recirculation line break inside primary containment, and the control rod drop accident. No increase in the probability of the evaluated accidents will result from crediting the proposed containment pressure because post-LOCA containment pressure does not represent an accident initiator but, rather, is an expected condition that will inherently exist in the containment after the pipe break inside containment.

The worst radiological consequences for the Pilgrim plant are associated with the design basis LOCA which is the double guillotine failure of the recirculation system piping. The radiological analysis of this event contained in FSAR Chapter 14 uses a TID–14844 source term and assumes a 1.5% per day leakage from the containment, which is greater than the maximum leakage allowed by the Technical Specifications. The results of this analysis are presented in Table 14.5–2 of the FSAR and indicate substantial margin when compared to 10 CFR Part 100 limits.

The radiological consequences of the design basis accident are not increased by taking credit for the post-LOCA suppression pool overpressure. Assuming containment integrity exists, the mechanism for increasing the consequences of the accident would be an increased leakage rate caused by an increase of the average differential pressure between primary and secondary containment during the accident response. However, the NPSH analysis performed for Pilgrim that includes post-LOCA containment pressure does not assume or require that the differential pressure between primary and secondary containment be maintained above the lower bounding minimum that exists due to thermal equilibrium conditions between the containment atmosphere and the suppression pool. Specifically, the containment pressure included in the ECCS pump NPSH analysis is inherently provided by the increase in wetwell vapor pressure and air/nitrogen partial pressure that exists due to equilibrium with increasing pool temperature with an accounting for containment initial conditions and leakage.

Inclusion of the post-LOCA containment pressure in the calculation of NPSH does not require that a higher containment pressure than would otherwise occur be purposely maintained, no requirement is incurred to delay operating containment heat removal equipment at the highest rate possible, no requirement is incurred to deliberately continue any condition of high containment pressure to maintain adequate NPSH, and no requirement is incurred for the purposeful addition of air/nitrogen into the containment to increase the available pressure.

The higher debris head losses that required the new NPSH evaluation are based on an updated analysis of LOCA-generated debris. The new debris analysis was performed in response to NRC Bulletin 96–03 using the guidance given in Regulatory Guide 1.82, Revision 2. The NRC guidance is used to ensure sufficient NPSH margin exists to accommodate the debris resulting from a LOCA. Using the proposed containment pressure limits included in this submittal, it is shown there is sufficient NPSH margin at all times following the bounding design basis accident.

Based on these reasons, the probability of accidents previously evaluated is not increased and the consequences of the design basis accident are not increased.

(2) Will crediting the proposed post-LOCA containment pressure create the possibility for new or different kinds of accidents?

As stated above, Chapter 14 of the FSAR contains the worst postulated accidents that the Pilgrim plant was evaluated for, which include the refueling accident, the main steam line break outside primary containment, the recirculation line break inside primary containment, and the control rod drop accident. New or different types of accidents are not created by including the containment pressure in NPSH analyses because post-LOCA containment pressure is an expected condition that will exist in the containment after the pipe break inside containment. The pressure included in the NPSH analysis is the minimum pressure that will exist due to thermal equilibrium conditions and must be considered as part of any accident analysis regardless of whether it is used in the evaluation of NPSH.

(3) Will crediting the proposed new limits for post-LOCA containment pressure in ECCS NPSH analyses involve a significant reduction in a margin of safety?

The integrity of the primary containment and the operation of the ECCS systems in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. In order for the ECCS pumps to meet their performance requirements, the NPSH available to the pumps throughout the accident response must meet their specific NPSH requirements. Excess NPSH margin will not improve the performance of the ECCS pumps because NPSH available must only meet NPSH requirements for the pump to operate on its pump curve and meet design expectations.

Including the proposed post-LOCA containment pressure in NPSH analyses increases the NPSH available to the ECCS pumps, but the methodology used includes only that pressure that will inherently exist due to thermal equilibrium between the containment atmosphere and the suppression pool because of the primary containment enclosure with an accounting for leakage. Post-accident containment pressure calculated in such a manner represents a conservative lower bound for the pressure that will be available. Therefore, it is expected the actual NPSH margin will exceed that calculated by these methods. The proposed pressure limits are enveloped at all times by the containment pressure calculated using the thermal equilibrium methodology. These methods for calculating NPSH available and NPSH margin were previously reviewed by the NRC for License Amendment 173.

The new debris analysis referenced in this submittal was done in accordance with Regulatory Guide 1.82, Revision 2. The LOCA debris analysis is considered conservative and bounding for all postulated accidents and transients. It is shown that, within the proposed containment pressure limits, there is sufficient NPSH margin at all times following the design basis accident to accommodate the debris head loss without affecting RHR [residual heat removal] or core spray pump performance.

Based on the above discussion, credit for the updated values of containment pressure in ECCS NPSH analyses does not involve a reduction in the margin of safety.

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

Attorney for licensee: W. S. Stowe, Esquire, Entergy Nuclear Generation Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Section Chief: James W. Clifford.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2 (ANO–1&2), Pope County, Arkansas; Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi; Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana; and Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: November 23, 1999.

Description of amendment request: The proposed amendments would incorporate the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into each facility's Technical Specifications (TS). Entergy Operations, Inc. (the licensee) is submitting these proposed amendments as a complete response to NRC Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

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:

I. The proposed change does not signficantly increase the probability or consequences of an accident previously evaluated.

The proposed changes do not result from a physical change to the facilities or impact plant operations. Neither do they impact the response of the facilities to an accident.

American Society of Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' reflects the most up-to-date method for accurately testing the efficiency of activated charcoal contained in engineered safety features (ESF) system adsorbers. Establishing ASTM D3803–1989 as the required method for laboratory testing of activated charcoal represents an upgrade from the current TS requirements. Using ASTM D3803–1989 methodology ensures the tested charcoal will perform in a manner consistent with the facility's licensing basis.

The proposed acceptance criterion values for charcoal efficiency were calculated using the equation specified in GL 99–02. As documented in GL 99–02, the NRC [Nuclear Regulatory Commission] found this equation acceptable for determining charcoal efficiency when using ASTM D3803–1989 as the test method.

Based on the above discussion, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not result from a physical change to the facilities or impact plant operations.

Establishing ASTM D3803–1989 as the method for performing laboratory testing of nuclear-grade activated charcoal does not involve a physical alteration to the facility or impact plant operations. Using ASTM D3803–1989 methodology ensures the tested charcoal will perform in a manner consistent with the facility's licensing basis.

The proposed acceptance criterion values for charcoal efficiency were calculated using the equation specified in GL 99–02. As documented in GL 99–02, the NRC found this equation acceptable for determining charcoal efficiency when using ASTM D3803–1989 as the test method.

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

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

The proposed changes do not result from a physical change to the facilities or impact plant operations. Neither do they impact the response of the facilities to an accident.

Safety-related air-cleaning units used in the ESF ventilation systems of nuclear power plants reduce the potential onsite and offsite consequences of a radiological accident by adsorbing radioiodine. To ensure the charcoal adsorbers used in these systems perform in a manner that is consistent with the facility's licensing basis, facility TS contain requirements to periodically test (in a laboratory) samples of charcoal taken from the air-cleaning units.

ASTM D3803–1989 reflects the most up-to-date method for accurately testing the efficiency of activated charcoal contained in ESF system adsorbers. Establishing ASTM D3803–1989 as the required method for laboratory testing of activated charcoal represents an upgrade from the current TS requirements and maintains the margin of safety by ensuring the tested charcoal performs in a manner consistent with the facility's licensing basis.

The proposed acceptance criterion values for charcoal efficiency were calculated using the equation specified in GL 99–02. As documented in GL 99–02, the NRC found this equation acceptable for determining charcoal efficiency when using ASTM D3803–1989 as the test method.

Based on the above discussion, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502; and Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 23, 1999.

Description of amendment request: The proposed amendments would make the following changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2) Technical Specifications (TSs): (1) For BVPS–1, surveillance requirement (SR) 4.8.1.1.2.b.3.b would be revised to reflect a narrower required diesel generator (DG) frequency band; an associated footnote would be deleted; associated Bases would be revised to reflect these TS changes. (2) For BVPS–2, SR 4.8.1.1.2.f would be revised to clarify that the DGs are only required to achieve a minimum frequency and voltage within the first 10 seconds of the related test, and that the stated voltage and frequency bands are requirements for steady state operation of the DGs; a footnote is also added to this SR. (3) Page formats are revised as needed to permit the addition or deletion of text.

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

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

For the Beaver Valley Power Station (BVPS) Unit No. 1 only, the proposed amendment will revise surveillance requirement (SR) 4.8.1.1.2.b.3.b.

Specifically, the required diesel generator (DG) frequency band specified in SR 4.8.1.1.2.b.3.b will be reduced. In addition, Footnote (6) pertaining to the DG frequency limits and associated Bases wording will be deleted.

For BVPS Unit No. 2 only, SR 4.8.1.1.2.f will be revised to clarify that the diesel generators are only required to achieve a minimum voltage and frequency in ≥ 10 seconds. The DGs are then required to obtain voltages and frequencies within the required bands during steady state operation. A new Footnote (8) will be added which modifies the stated voltage and frequency values in the proposed SR 4.8.1.1.2.f.1. This footnote will require the voltage and frequency values be appropriately increased to account for measurement uncertainties.

Page format will be revised as necessary to permit incorporation and deletion of text. These format changes include the addition or deletion of Technical Specification pages as required.

The DGs are used to support mitigation of the consequences of a design basis accident (DBA); however, they are not considered the initiator of any previously analyzed DBA described in the Updated Final Safety Analysis Report (UFSAR). The proposed amendment does not impact any of the offsite AC distribution system; therefore, the probability of a loss of offsite power event is not increased.

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

The proposed reduction in the DG output frequency limits (for BVPS Unit No. 1 only) will continue to protect engineered safety feature (ESF) pumps from runout conditions and ESF pump motors from operating in an unanalyzed condition. The revised frequency limits have no adverse effect on the diesel generator operability. The revised DG output frequency limits do not increase the consequences of a design basis accident; this proposed change ensures that equipment will perform its intended function. This change is intended to prevent the DG from being loaded beyond analyzed loading limits and protect ESF equipment. The revised surveillance requirements being applied to operating limits will provide greater assurance that increased performance requirements are not imposed on ESF equipment.

The proposed deletion of Footnote (6) (for BVPS Unit No. 1 only) removes the ability to evaluate the DG frequency response. The proposed wording is more restrictive in that the DG frequency response will be required to be demonstrated regardless of the amount of DG loading. The ability of the DGs to maintain the required output frequency as required to meet accident analysis assumptions will continue to be demonstrated on a periodic basis. The proposed deletion of the Bases wording pertaining to Footnote (6) is administrative in nature and does not affect plant safety. This change removes guidance information on how to conduct the engineering evaluation that will no longer be applicable following DG governor modifications.

The proposed revision to SR 4.8.1.1.2.f (for BVPS Unit No. 2 only) will continue to require that both DGs start simultaneously to confirm that there is not a cross-tie that could render both DGs incapable of performing their required functions. The proposed revision to SR 4.8.1.1.2.f will continue to require that each DG obtain the minimum conditions to accept load in the time frame assumed in the accident analysis. In addition, the proposed wording of SR 4.8.1.1.2.f will continue to require that each DG obtain the required steady state voltage and frequency values.

The proposed addition of Footnote (8) (for BVPS Unit No. 2 only) is administrative in nature and does not affect plant safety. The proposed footnote provides information that the values for voltage and frequency need to be increased to account for measurement uncertainties.

The revision to page format as necessary to permit incorporation and deletion of text is editorial in nature and does not affect plant safety.

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

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

The proposed revisions do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions have no adverse impact on the DBAs previously evaluated in the UFSAR. The proposed revisions will continue to assure that the DGs are available and fully operable to perform their intended safety function of providing sufficient electrical power to ESF equipment following a DBA and a loss of offsite power. New failure modes are not introduced as a result of the proposed revisions to the DG surveillance requirements. The proposed revision to the required DG frequency range will continue to prevent ESF motors and pumps from being subjected to overfrequency conditions which could reduce the life of the equipment. The proposed changes do not affect the probability of malfunction of a DG or its connected emergency AC power system.

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

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

The margin of safety is not significantly reduced as a result of the proposed revisions. The margin of safety depends on the maintenance of specific operating parameters within design limits.

The BVPS Unit No. 1 DG reliability and performance during a loss of offsite power and a DBA are enhanced by the proposed revision to SR 4.8.1.1.2.b.3.b. This proposed revision (for BVPS Unit No. 1 only) will ensure that the maximum calculated DG loading does not exceed the UFSAR limit of 2745 kW. The proposed revision to SR 4.8.1.1.2.b.3.b for DG operating frequency limits continues to protect ESF equipment from overfrequency conditions. ESF equipment will continue to function, as assumed in the safety analysis, to ensure that fuel, reactor coolant system and containment design limits are not exceeded.

The proposed revision to SR 4.8.1.1.2.f (for BVPS Unit No. 2 only) will continue to require that both DGs start simultaneously to confirm that there is not a cross-tie that could render both DGs incapable of performing their required functions. The proposed revision to SR 4.8.1.1.2.f will continue to require that each DG obtain the minimum conditions to accept load in the time frame assumed in the accident analysis. In addition, the proposed wording of SR 4.8.1.1.2.f will continue to require that each DG obtain the required steady state voltage and frequency values.

The remaining changes are either administrative or editorial in nature and do not affect plant safety.

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

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

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

NRC Section Chief: Marsha Gamberoni.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: April 15, 1999, as supplemented on December 22, 1999.

Description of amendment request: The proposed amendment would change the Technical Specification as described below:

Page 1.0–3 Clarification would be added to the definition of Secondary Containment Integrity.

Page 1.0–4 The definition of facility description and safety analysis report (FDSAR) would be expanded.

Page 2.3–3 The Bases section would be separated from this page which is the last page of the specification.

Page 2.3–4 Two paragraphs, which should have been deleted in an earlier revision, would be deleted and subsequent pagination would be affected. Two paragraphs would be moved from the end of the bases to that location. An unrelated wording change would also be made.

Page 2.3–7 In addition to pagination, a sentence would be added about the relays involved in undervoltage situations.

Page 3.4–1 The phrase ''(see Note below)'' would be deleted as unnecessary and two lines from the top of page 3.4–2 would be included as ''c.''

Page 3.4–2 Two lines would be moved to the prior page and designated ''c.''

Page 3.5–7 LCO statement of 36 hours would be deleted from Specification 3.5.B.6.a.3 because it is inconsistent with Specification 3.5.B.7.

Page 3.5–9 A bases statement would be added about administrative control over non-automatic primary containment isolation valves.

Page 3.5–11 A bases statement would be added about the use of the trunion room door.

Page 3.7–1 The phrase ''shutdown position'' would be corrected to ''shutdown condition.''

Page 3.17–1 The phrase ''the control room HVAC system'' would be corrected to ''one control room HVAC system.''

Page 4.5–13 The word ''off'' would be changed to ''on.''

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

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

The proposed change are relatively minor in nature and are proposed to enhance clarity and understanding. None of the changes have any impact on safety and there is no change to an operating parameter of any system, component or structure. Accordingly, the proposed changes do not affect any accident precursors. Therefore, the probability of an accident previously evaluated is not increased. The proposed TS change will assure the ability of systems to perform their intended function. Therefore, the proposed changes will not involve a significant increase in the consequences of an accident previously evaluated. Therefore, the probability of occurrence or the consequences of an accident previously evaluated in the Safety Analysis Report (SAR) will not increase as a result of these changes.

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

The proposed changes are relatively minor in nature and are proposed to enhance clarity and understanding. None of the changes have any impact on safety and there is no change to an operating parameter of any system, component or structure. The proposed changes do not involve placing systems in new configurations or operating systems in a different manner that could result in a new or different kind of accident. Therefore, the proposed activity does not create the possibility for a new or different kind of accident from any previously identified in the SAR.

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 changes are primarily administrative and are proposed to enhance clarity and understanding. They do not modify an operating parameter of any system, component or structure. They do not adversely affect the performance characteristics of systems nor do they affect the ability of systems to perform their intended function. Therefore, the proposed changes do not involve a significant reduction in the 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: M. Gamberoni, Acting.

PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station (LGS), Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: December 15, 1999.

Description of amendment request: The proposed changes will revise the LGS Technical Specifications (TSs) to remove TS Table 3.6.3–1, ‘‘Primary Containment Isolation Valves,’’ and references to the table, from the TSs and relocate the information from the TS table to the Technical Requirements Manual, a licensee-controlled document.

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

The probability of occurrence of a previously evaluated accident is not increased because containment isolation is not an accident initiator and the proposed changes do not impact any accident initiating conditions. The consequences of an accident previously evaluated are not increased because the proposed changes do not impact the ability of containment to restrict the release of any fission product radioactivity to the environment. The proposed change to remove the primary containment isolation valve table from TS and relocate the information to an administratively controlled document, and to revise the wording in TS to reflect this change, will have no impact on any safety related structures, systems or components. The Technical Specification requirements for the primary containment isolation valves will not be changed. In addition, the details of the table are not being changed, only relocated to a different controlling document. The proposed changes simplify the Technical Specifications, meet the regulatory requirements for control of containment isolation, and are consistent with the guidance provided in Generic Letter 91–08. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature and do not result in physical alterations or changes in the method by which any safety related system performs its intended function(s). The proposed changes do not impact any safety analysis assumptions. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to remove the primary containment isolation valve table from TS and relocate the information to an administratively controlled document, and to revise the wording in TS to reflect this change, do not alter the Technical Specifications requirements for containment integrity and containment isolation and will not adversely affect the containment isolation capability. The licensee controlled document will be maintained under the requirements of TS Administrative Controls Section 6.0 and the provisions of 10CFR50.59. In addition, the proposed changes do not impact any safety analysis assumptions. 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: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC. Section Chief: James W. Clifford.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: February 9, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Limiting Condition for Operation (LCO) 3.8.2.1. The proposed change would add two new Action Statements for operating conditions where a Class-1E battery's electrolyte temperature is below the minimum limit specified in TS Surveillance Requirement 4.8.2.1.b.3.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC). The Class-1E batteries will continue to function as designed. The Class-1E battery system is designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. The proposed TS LCO Action Statements will continue to ensure that the Class-1E batteries are capable of performing their required safety functions while providing a sufficiently conservative period of continued plant operation. In addition, this proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety, since the manner in which the Class-1E battery system is operated is not affected by these proposed changes. The operating limits specified in the proposed TS LCO ensure that the battery's safety functions will be accomplished. Therefore, the proposed TS changes would not result in the increase of the consequences of an accident previously evaluated, nor do they involve an increase in the probability 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 TS changes do not involve any physical changes to the design of any plant SSC. The design and operation of the Class-1E battery system is not changed from that currently described in the UFSAR [Updated Final Safety Analysis Report], only the allocation of battery design margin would be temporarily affected by the proposed TS LCO. The Class-1E battery system will continue to function as designed to mitigate the consequences of an accident. Establishing a 31 day period where a Class-1E battery would be considered operable, with electrolyte temperature at or above 65°F and Category A and Category B limits met as appropriate, does not permit plant operation in a configuration that would create a different type of malfunction to the Class-1E batteries than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in the UFSAR regarding the safety related functions of the Class-1E batteries or their support systems.

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

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

The proposed changes contained in this submittal would implement TS requirements that either: (1) Permit continued plant operation when the safety function of the Class-1E batteries can be performed; or (2) conservatively require placing the plant in a safe shutdown condition. A Class-1E battery operating within Category A and Category B limits as appropriate and a 65°F battery electrolyte temperature (for a limited 31 day period) will still perform its safety-related functions. Temporary allocation of battery capacity margins in compensation of degraded operating conditions (low specific gravity) is already permitted by the Hope Creek TS (for a 31 day period). The ability of the Class-1E batteries to independently supply their required loads for four hours without support from battery chargers is not adversely affected by these proposed changes. Therefore, the proposed TS 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit–N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: January 19, 2000 (TSCR 217).

Description of amendment request: The proposed amendments would revise Technical Specification 15.4.4 to clarify that a different containment tendon may be designated as a control tendon providing that the new control tendon had not previously been physically changed.

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

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

The proposed change does not involve a change to structures, systems, or components which would affect the probability or consequences of an accident previously evaluated in the PBNP Final Safety Analyses Report (FSAR). The containment tendons are components integral to maintaining the containment pressure boundary under post accident conditions. Neither the tendons nor the containment tendon testing process are accident initiators. The proposed change simply clarifies the Technical Specifications regarding the selection of control tendons used to develop a tendon relaxation history and correlate observed test data. The proposed change does not affect reactor operations or accident analysis and has no significant radiological consequences. Therefore, this change will not create a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents. This change clarifies the Technical Specifications regarding the selection of control tendons used to develop a history and correlate observed test data. Except for the method of selecting the control tendon, the methods for performing the actual tendon surveillances are not changed. No new accident modes are created by selecting the control tendons. No safety-related equipment or safety functions are altered as a result of this change. Selecting a control tendon has no influence on, nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. Therefore, the proposed change will not create the possibility of a new or different kind of accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The proposed change affects only the selection of control tendons used to develop a history and correlate observed test data. Except for the method of selecting the control tendons, the methods for performing the actual tests are not changed. The proposed change is based on NRC accepted provisions contained in Regulatory Guide 1.35, Revision 3. Furthermore, the proposed change will not reduce the availability of systems associated with containment integrity when they are required to mitigate accident conditions. Therefore, the proposed change will not create 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: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Claudia M. Craig.

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, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina.

Date of application for amendment: July 9, 1999, as supplemented on January 19, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) 3⁄4.2.2, ''Heat Flux Hot Channel Factor—$F_Q(Z)$,'' TS 3⁄4.2.3, ''RCS Flow Rate And Nuclear Enthalpy Rise Hot Channel Factor,'' TS 3⁄4.2.5, ''DNB Parameters,'' an associated note in TS Table 2.2–1, and associated Bases. Specifically, the proposed amendment would: (1) Remove the allowance for reduced power operation for reduced Reactor Coolant System (RCS) flow rate conditions; (2) separate the requirements for F delta H and RCS flow rate in the format prescribed by NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995; and (3) implement the guidance of NUREG–1431, Revision 1, and NRC Generic Letter 88–16, dated October 4, 1988, for TS 3⁄4.2.2 and TS 3⁄4.2.3 and associated Bases by removing cycle-specific parameters and placing that information into the Core Operating Limits Report.

Date of issuance: February 24, 2000.

Effective date: February 24, 2000.

Amendment No. 95.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** August 11, 1999 (64 FR 43765).

The January 19, 2000, submittal contained clarifying information only, and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2000.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: July 9, 1999.

Brief description of amendment: This amendment revises the Technical Specifications (TS) by relocating several instrumentation TS to plant procedures.

Date of issuance: February 24, 2000.

Effective date: February 24, 2000.

Amendment No. 96.

Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** August 11, 1999 (64 FR 43766).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2000.

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: July 30, 1999, as supplemented by letter dated December 2, 1999.

Brief description of amendment: By application dated July 30, 1999, as supplemented by letter dated December 2, 1999, Entergy Operations, Inc. (the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. NPF–47) for the River Bend Station, Unit 1. The proposed change, more commonly referred to as ''power uprate,'' would revise the TSs and the operating license to increase the current licensed power of 2894 megawatts thermal (MW_{th}) to the uprated power level of 3039 MW_{th}, an increase of 5 percent. Included in the power uprate license amendment application was a request to increase the main steam safety and relief valves (S/RV) safety mode/function setpoint tolerance defined in Surveillance Requirement (SR) 3.4.4.1 from +0/−2 percent to ±3 percent.

This amendment approves, prior-to the issuance of the power uprate license amendment, a portion of the S/RV setpoint tolerance change requested. The change increases the safety function lift setpoint tolerances for the S/RVs listed in SR 3.4.4.1 from the current +0/−2 percent of the safety function lift setpoint to +0/−3 percent (*i.e.,* a partial 3 percent tolerance). The remaining (''+3 percent'') portion of the proposed setpoint tolerance change will be reviewed in conjunction with approval for the power uprate.

Date of issuance: February 9, 2000.

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

Amendment No.: 109.

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

Date of initial notice in **Federal Register:** November 17, 1999.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 23, 1998.

Brief description of amendment: Modification of Limiting Condition for Operation for the chlorine detection system and correction of typographical error in Table 3.3–4.

Date of issuance: February 11, 2000.

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

Amendment No.: 156.

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

Date of initial notice in **Federal Register:** February 24, 1999 (64 FR 9190).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 11, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 3, 1999.

Brief description of amendment: The amendment revises Final Safety Analysis Report, Section 9.5.4.1. The revision changes this section to explicitly list the Waterford Steam Electric Station, Unit 3 (Waterford 3) deviations from American National Standards Institute (ANSI) Standard N195–1976.

Date of issuance: February 15, 2000.

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

Amendment No.: 157.

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

Date of initial notice in **Federal Register:** November 17, 1999 (64 FR 62713).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 15, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 24, 1998, as supplemented January 6, 1999.

Brief description of amendments: These amendments change the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2) Technical Specifications (TSs) to ensure that Emergency Diesel Generator (EDG) requirements contained in Technical Specification 3/4.8.1 for both units are consistent with assumptions contained in design analyses and requirements of plant procedures. Revisions to TS 3/4.8.1, ''A.C. Sources,'' contained in these amendments provide more conservative limiting conditions for operation and surveillance requirements that affect EDG fuel oil storage volume, EDG load rejection and overspeed testing, and EDG operating frequency requirements. The applicable bases for each unit are also refined, as necessary, to strengthen the explanations regarding EDG fuel oil storage systems and provide the EDG overspeed in terms of frequency (Hertz) and speed (Revolutions Per Minute).

Date of issuance: February 11, 2000.

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

Amendment Nos.: 227 and 105.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 27, 1999, (64 FR 4154). The January 6, 1999, letter requested a 60-day implementation period. This letter did not change the initial proposed no significant hazards consideration determination or expand the amendments beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 11, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 27, 1999.

Brief description of amendment: This amendment eliminates TS 6.4, ''Training,'' and relocates TS 6.5.2.8, ''Audits,'' and TS 6.10, ''Record Retention,'' to the USAR Chapter 17 Quality Assurance Program. Additionally, the record keeping requirements of TS 6.14, ''Process Control Program,'' and TS 6.15, ''Offsite Dose Calculation Manual,'' are also being relocated to the USAR Chapter 17 Quality Assurance Program. Finally, an editorial change has been made to TS 6.8, ''Procedures and Programs.''

Date of issuance: February 14, 2000.

Effective date: Immediately, to be implemented within 120 days.

Amendment No.: 235.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 8, 1999 (64 FR 48863).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 8, 1999

Brief description of amendment: This amendment relocates Technical Specification (TS) 6.5.1, Station Review Board, and TS 6.5.2, Company Nuclear Review Board, to the Davis-Besse Nuclear Power Station Updated Safety Analysis Report Chapter 17.2, Quality Assurance During the Operations Phase. These changes are consistent with the recommendations in NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance,'' dated December 12, 1995.

Date of issuance: February 14, 2000.

Effective date: Immediately, to be implemented within 120 days.

Amendment No.: 236.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 15, 1999 (64 FR 70087).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 1, 1999, as supplemented September 25, 1999.

Brief description of amendments: These amendments revise the St. Lucie, Units 1 and 2, Technical Specifications (TS) 3.5.2 to allow up to 7 days to restore an inoperable Low Pressure Safety Injection train to operable status.

Date of Issuance: February 15, 2000.

Effective Date: February 15, 2000.

Amendment Nos.: 164 and 106.

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

Date of initial notice in **Federal Register:** June 30, 1999 (64 FR 35206). The supplemental September 25, 1999, letter provided additional information that did not expand the scope of the amendment request beyond the initial notice or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 15, 2000.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: November 17, 1999 *Brief description of amendment:* The amendment revised the technical specification (TS) surveillance testing of the safety-related ventilation system charcoal to meet the actions requested in Generic Letter 99–02, ''Laboratory testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

Date of Issuance: February 17, 2000.

Effective Date: February 17, 2000.

Amendment No.: 107.

Facility Operating License No. NPF–16: Amendment revised the TSs.

Date of initial notice in **Federal Register:** January 12, 2000 (65 FR 1923).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 2000.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: December 28, 1999.

Brief description of amendment: The amendment revised Technical Specifications Table 4.4.6.1.3–1, ''Reactor Vessel Material Surveillance Program—Withdrawal Schedule.'' The revised requirement permits the withdrawal of surveillance capsule number 1 at 8 effective full-power years (EFPY) instead of the original 10 EFPY.

Date of issuance: February 15, 2000.

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

Amendment No.: 90.

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

Date of initial notice in **Federal Register:** January 14, 2000 (65 FR 2443).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 15, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: December 16, 1999, as supplemented January 21, 2000.

Brief description of amendment: The amendment revises the Technical Specification (TS) Safety Limit Minimum Critical Power Ratio (SLMCPR) values for two recirculation pump and single-loop operation, deletes cycle specific footnotes, updates the single-loop operation Average Planar Heat Generation rate limiting values, corrects a typographical error, and deletes an obsolete reference to Siemens fuel.

Date of issuance: February 16, 2000.

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

Amendment No.: 109.

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

Date of initial notice in **Federal Register:** December 29, 1999 (64 FR 73094).

The January 21, 2000, submittal provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 18, 1998.

Brief description of amendments: The amendments revise Bases 3/4.6.2.1, ''Containment Spray System,'' of the current Technical Specifications (TSs) and Bases 3.6.6, ''Containment Spray and Cooling Systems,'' of the improved TSs, to clarify that containment spray is not required to be actuated during recirculation, but may be actuated at the discretion of the Technical Support Center. Additionally, the Bases are clarified to state that the ability to spray containment using the residual heat removal (RHR) system is demonstrated by opening the RHR Spray Ring Cross Connect Valve 9003A or B. The Bases are clarified to state that flow to the spray headers can be established with only one operable RHR pump by closing the cold leg discharge valve 8809A or B.

Date of issuance: February 9, 2000.

Effective date: February 9, 2000.

Amendment Nos.: Unit 1—139 ; Unit 2—139.

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

Date of initial notice in **Federal Register**: August 26, 1998 (63 FR 45527).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 10, 1998, as supplemented by letter dated November 24, 1999.

Brief description of amendments: These amendments revise Technical Specification (TS) 3/4.3.2, Table 3.3–5, ''Engineered Safety Features Response Times,'' of the current TS to add the response times for closure of the main feedwater regulating valves (MFRVs) and MFRV bypass valves, and trip of the main feedwater pumps (MFWPs). The change would also revise TS 3/4.7.1.7 to add a limiting condition for operation, actions, and surveillance requirements for the MFWP turbine stop valves, and revise the TS 3/4.7.1.7 actions and surveillance requirements for the MFRVs, MFRV bypass valves, and main feedwater isolation valves to be consistent with the NUREG–1431 requirements. Also, the amendments revise Section 3.7.3 and its associated bases of the improved Technical Specifications (ITS).

Date of issuance: February 22, 2000.

Effective date: February 22, 2000, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1—140; Unit 2—140.

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

Date of initial notice in **Federal Register**: October 7, 1998 (63 FR 53954). The November 24, 1999, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2000.

No significant hazards consideration comments received: No.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: November 24, 1999, as supplemented January 13, 2000.

Brief description of amendment: The amendment revises the Technical Specifications to remove the requirement for partial stroking of the main steam isolation valves twice-per-week.

Date of issuance: February 24, 2000.

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

Amendment No.: 260.

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

Date of initial notice in **Federal Register**: December 29, 1999 (64 FR 73095) The letter of January 13, 2000, provided supplemental information that did not affect the initial proposed no significant hazard consideration determination of the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24. 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 11, 1999 (PCN–499), as supplemented November 29, 1999.

Brief description of amendments: The amendments revise Technical Specification 3.7.6 to change the minimum inventory of water maintained in the condensate storage tank (T–120) from 280,000 gallons to 360,000 gallons during plant operation Modes 1, 2, and 3.

Date of issuance: February 22, 2000.

Effective date: February 22, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—162; Unit 3—153.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 18, 2000 (65 FR 2648). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 8, 1999 (PCN–495).

Brief description of amendments: The amendments modify the Technical Specifications to (1) reflect that charging flow is not required to mitigate the effects of design-basis small-break loss-of-coolant accidents (SBLOCAs), (2) increase the maximum as-found lift pressure positive tolerance of main steam safety valves from +1 percent to +2 percent of the setting, and (3) list the ABB Combustion Engineering Supplement 2 SBLOCA evaluation model as an acceptable method for determining linear heat rate.

Date of issuance: February 22, 2000.

Effective date: February 22, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—163; Unit 3—154

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 30, 1999 (64 FR 35210)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 2, 1998 (PCN–482), as supplemented December 13, 1999.

Brief description of amendments: The amendments revise Technical Specification 3.7.5 to add a note that states: The steam driven AFW [auxiliary feedwater] pump is OPERABLE when running and controlled manually to support plant start-ups, plant shut-downs, and AFW pump and valve testing.

Date of issuance: February 23, 2000.

Effective date: February 23, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—164; Unit 3—155.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register** January 19, 2000 (65 FR 2991), as corrected January 26, 2000 (65 FR 4265)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 2000.

No significant hazards consideration comments received: No

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

Date of application for amendments: March 19, 1999.

Brief description of amendments: These amendments relocate Technical Specification Section 3/4.8.3, ''Electrical Equipment Protective Devices,'' and the associated bases to the Technical Requirements Manual.

Date of issuance: February 22, 2000.

Effective date: As of date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 250 and 241.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register**: April 21, 1999 (64 FR 19566).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 1st day of March 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–5477 Filed 3–7–00; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–24323; File no. 812–11850]

Seligman Portfolios, Inc., et al.

February 29, 2000.

AGENCY: Securities and Exchange Commission (the ''SEC'' or the ''Commission'').

ACTION: Notice of application for an order under Section 6(c) of the Investment Company Act of 1940 (the ''1940 Act'') for exemptions from the provisions of Sections 9(a), 13(a), 15(a) and 15(b) of the 1940 Act and Rules 6e–2(b)(15) and 6e–3(T)(b)(15) and 6e–3(T)(b)(15) thereunder.

SUMMARY OF APPLICATION: Applicants seek an order to permit shares of Seligman Portfolios, Inc. and shares of any other open-end investment company that is designed to fund insurance products and for which J. & W. Seligman & Co. Inc., or any of its affiliates, may serve, now or in the future, as investment adviser, administrator, manager, principal underwriter or sponsor (Seligman Portfolios, Inc. and such other investment companies hereinafter referred to collectively, as ''Insurance Products Funds'') to be offered to, sold to and held by (a) variable annuity and variable life insurance separate accounts of both affiliated and unaffiliated life insurance companies; and (2) qualified pension and retirement plans outside of the separate account context.

APPLICANTS: Seligman Portfolios, Inc. (''Seligman Portfolios'') and J. & W. Seligman & Co. Inc. (''Seligman'').

FILING DATE: The application was filed on November 16, 1999, and amended and restarted on January 27, 2000, and February 25, 2000.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing on this application by writing to the Secretary of the SEC and serving Applicants with a copy of the request, in person or by mail. Hearing requests must be received by the Commission by 5:30 p.m. on March 23, 2000, and accompanied by proof of service on the Applicants in the form of an affidavit or,