concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff person named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Dr. John T. Larkins (telephone: 301/415–7360) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any changes in schedule, etc., that may have occurred.

Dated: October 12, 2000.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–26758 Filed 10–17–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the Subcommittee on Plant Systems; Notice of Meeting

The ACRS Subcommittee on Plant Systems will hold a meeting on October 31, 2000, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Tuesday, October 31, 2000—8:30 a.m. until 12:00 noon.*

The Subcommittee will discuss the safety evaluation reports on the topical reports for ABB/CE and Siemens Digital I&C Applications. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman and written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Amarjit Singh (telephone 301/415–6899) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: October 6, 2000.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 00–26759 Filed 10–17–00; 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 September 25, 2000, through October 6, 2000. The last biweekly notice was published on October 4, 2000.

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's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC through September 22, 2000. The NRC is relocating its Public Document Room to the NRC's headquarters building. Effective September 26, 2000, documents may be examined at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 17, 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, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly available records will be accessible 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible 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: August 9, 2000.

Description of amendment request: The proposed amendment revises Sections 6.5.3 and 6.5.4 of the Technical Specifications to eliminate reference to the Independent Onsite Safety Review

Group (IOSRG) and to redefine the performance of the IOSRG function by the nuclear quality assurance organization.

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 facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed changes do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR [Updated Final Safety Analysis Report] transient analyses. No Technical Specification Limiting Condition for Operation, Action Statement, or Surveillance Requirement is affected by any of the proposed changes. The proposed changes do not alter the design, function, or operation of any plant component. Therefore, the proposed amendment does not affect the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes do not affect assumptions contained in the plant safety analyses, the physical design and/or modes of plant operation defined in the plant operating license, or Technical Specifications that preserve safety analysis assumptions. The proposed changes do not introduce a new mode of plant operation or surveillance requirement, nor involve a physical modification to the plant. The proposed changes do not alter the design, function, or operation of any plant components. Therefore, the proposed amendment does not affect the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR transient analyses. No Technical Specification Limiting Condition for Operation, Action Statement, or Surveillance Requirement is affected. Therefore, the proposed amendment does not reduce the margin of safety.

Based upon the analysis provided herein [the licensee's August 9, 2000 application], the proposed changes will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a reduction in a margin of safety. The performance of safety assessment and the IOSRG functions by a single qualified organization will lead to efficiencies in the performance of both functions. The training and qualification of the personnel performing the IOSRG functions will be unchanged from the current requirements. Therefore, the proposed changes meet the requirements of 10 CFR 50.92(c) and involve no significant hazards consideration.

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

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: August 9, 2000.

Description of amendment request: The proposed amendment revises the Three Mile Island Nuclear Station, Unit 1 (TMI–1), Updated Final Safety Analysis Report (UFSAR), Section 14.1.2.10, ''Steam Generator Tube Failure Analysis,'' to include the dose resulting from the postulated post-accident steam release through the main steam safety valves. The revised dose for the TMI–1 steam generator tube failure analysis would be increased above the values previously reviewed and approved by the NRC, but would continue to be below the limits in Title 10 of the Code of Federal Regulations (10 CFR) Part 100. The proposed change to the UFSAR modifies the existing analysis to account for release of radioactivity to the atmosphere for the postulated tube rupture analysis. The existing dose calculations do not account for this release. Editorial and grammatical corrections are also made.

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 facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. This change has no effect on structures, systems or components prior to the postulated steam generator tube failure accident or any other accident. The proposed change corrects the existing UFSAR Steam Generator Tube Failure accident analysis to account for the release to atmosphere through the main steam safety valves (MSSVs). The resulting revised radiological consequences for the postulated Steam Generator Tube Failure accident remain well below the 10 CFR 100 limits.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. This change has no impact on any plant structures systems or components. The only impact is the revised radiological consequences of the Steam Generator Tube Failure accident analysis to account for the release to atmosphere through the MSSVs. This change only corrects the existing TMI Unit 1 UFSAR.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. No change to any plant structure, system or component is being made or proposed by this change. This change does not involve any change to safety system setpoints for operation. The revised radiological consequences of the Steam Generator Tube Failure accident analysis remain well below 10 CFR 100 limits.

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.

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

Date of amendments request: September 14, 2000.

Description of amendments request: The proposed amendment revises the Unit 1 and Unit 2 heatup curves (Technical Specification Figures 3.4.3–1) and Unit 1 and Unit 2 cooldown curves (Technical Specification Figures 3.4.3–2) to increase the allowable heatup and cooldown rates. Use of stress intensity factor K_{IC}, permitted by American Society of Mechanical Engineers (ASME) Code Case N–640, made it possible to increase the heatup and cooldown rates without changing existing pressure-temperature (P–T) limits. The existing P–T limits were approved previously. Application of Code Case N–640 to generate P–T curves is not currently permitted by the regulations. Therefore, pursuant to 10 CFR 50.12, a separate request for an exemption to use Code Case N–640 was submitted in a letter dated September 14, 2000.

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

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

In accordance with 10 CFR Part 50, Appendix G, the Calvert Cliffs pressure/temperature (P–T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials were developed using the methods of linear elastic fracture mechanics and the guidance found in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G. The Calvert Cliffs P–T limits are based on fluence level. The fluence levels are determined in the same manner as the pressurized thermal shock (PTS) screening criteria defined in 10 CFR 50.61 for the critical elements. Methods described in the Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, are used to predict the embrittlement effect of neutron irradiation on reactor vessel materials. Regulatory Guide 1.99 defines embrittlement effect in terms of adjusted reference temperatures, which depends on the material property of the PTS critical elements.

The proposed higher heatup and cooldown rates for the Technical Specification P–T limits were made possible by the ASME Code Case N–640 which permits use of reference stress intensity factor K_{IC}, in place of K_{IA}. Use of K_{IC}, for the maximum stress intensity factor that will not lead to failure, is the correct value to use. Although conservative in terms of developing P–T limits, use of K_{IA} results in a very restrictive heatup and cooldown rate that challenges plant safety. To bound the existing LTOP [low-temperature overpressure protection] enable temperatures, while increasing the heatup and cooldown rates, the criteria described in ASME Section XI Code Case 514 is used. Code Case 514 is listed in Regulatory Guide 1.147 as acceptable to the Nuclear Regulatory Commission (NRC) for this application. With the new higher heatup and cooldown rates, the underlying intent of the 10 CFR Part 50, Appendix G, requirement for adequate margin to prevent brittle failure of the reactor coolant pressure boundary materials is maintained. Additionally, since the cooldown rates are not changed above 300° F, the safety analyses and dose consequences in the Updated Final Safety Analysis Report are not affected.

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

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

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

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

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

As discussed above, although conservative in terms of developing P–T limits, use of K_{IA} results in a very restrictive heatup and cooldown rate that challenges plant safety. The insignificant margin reduction in P–T limits is more than compensated by the safety benefits that are realized in terms of plant component integrity as a result of the higher heatup and cooldown rates. With the proposed change, the underlying intent of the 10 CFR Part 50, Appendix G, requirement for adequate margin to prevent brittle failure of the reactor coolant pressure boundary materials is maintained, and there is a net gain in overall plant safety margin.

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

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

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

Date of amendments request: September 14, 2000.

Description of amendments request: The proposed amendment adds two analytical methods to the list of approved core operating limits analytical methods in the Technical Specifications (TSs) for Calvert Cliffs, Unit Nos. 1 & 2. In a letter dated March 16, 2000, from Mr. S. A. Richards, NRC to Mr. I. C. Rickard, ABB Combustion Engineering, the Nuclear Regulatory Commission approved the Topical Report CENPD–387–P, ''ABB Critical Heat Flux Correlations for [pressurized-water reactor] PWR Fuel'' for referencing in licensing applications for Asea Brown Boveri, Inc. Combustion Engineering, Inc. (ABB–CE) plants.

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

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

The proposed change allows the use of the ABB–NV and ABB–TV CHF [critical heat flux] correlations in the thermal hydraulic analysis for Calvert Cliffs Nuclear Power Plant. The ABB–NV is used for a non-mixing vane fuel assembly and the ABB–TV correlations are used for Turbo mixing vane fuel assembly. The CHF correlations determine the departure from nucleate boiling ratio (DNBR). The specified acceptable fuel design limit for DNBR will change for ABB–NV and ABB–TV. The use of the ABB–NV and/or ABB–TV correlations with the appropriate DNBR limit provides additional operating margin for those analyses that presently use the CE–1 correlation.

The use of a different CHF correlation will not increase the probability of an accident because the plant systems will not be operated outside of design limits, the plant equipment will not be operated in a different manner, and system interfaces will not change.

As Turbo fuel is introduced to reactor, transistion cores will exist in which Turbo mixing vane grid fuel assemblies are co-residents with non-mixing vane grid fuel assemblies. The grid hydraulic loss coefficient in the Turbo grids is greater than the grid hydraulic loss coefficient for the non-mixing grids. The flow diversion that will result does not increase the probability of an accident previously evaluated because assembly flow has no impact on accident initiators, and because plant systems will not be operated outside of design limits, plant equipment will not be operated in a different manner, and system interfaces will not change.

The change in the CHF correlation was the subject of Topical Report CENPD–387–P–A, which was reviewed and approved by the NRC. The use of a different CHF correlation will not increase the consequences of an accident because Limiting Conditions [for] Operation (LOCs) will continue to restrict operation to within the regions that provide acceptable results, and Reactor Protection System (RPS) trip setpoints will plant transients so that the consequences of accidents will be acceptable.

The transition cores that will exist as Turbo fuel is introduced to the reactor will not increase the consequences of an accident. The TORC code accurately predicts the flow conditions in adjacent fuel bundles that contain grids with different designs and coefficients. The flow diversion will be compensated for by DNBR margin gains. Operation within the LOCs and RPS setpoints will continue to restrict plant transients so that consequences of accidents will be acceptable.

Therefore, the proposed TS 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 [kind] of accident from any accident previously evaluated.

The proposed change does not add any new equipment, modify any interfaces with any existing equipment, alter the equipment's function or change the method of operating the equipment. The proposed change does not alter plant conditions in a manner that could affect other plant components. The proposed change does not cause any existing equipment to become an accident initiator. The Turbo grid design does not introduce features that could 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.

Safety Limits ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. One of the safety limits that accomplishes this is the DNBR limit. The CHF correlations that have been approved for ABB–NV and ABB–TV result in a DNBR limit that provides a 95% probability, at a 95% confidence, that the hot fuel rod in the core will not experience departure from nucleate boiling. The RPS in combination with the LCOs, will continue to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure and thermal power level that would result in a violation of the Safety Limits.

Therefore the margin of safety is not significantly reduced by this proposed change.

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

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

Date of amendments request: September 14, 2000.

Description of amendments request: Calvert Cliffs Nuclear Power Plant, Inc. (CCNPPI) proposed an amendment to incorporate changes described below into the Technical Specifications (TSs) for Calvert Cliffs Units 1 and 2.

On September 9, 1996, a final rule amending 10 CFR 50.55a was issued requiring owners to implement, by September 9, 2001, the requirements of the 1992 Edition through the 1992 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsections IWE and IWL, as modified and supplemented by 10 CFR 50.55a. CCNPPI has developed a program to effect the implementation of Subsections IWE and IWL. This submittal requests a license amendment in support of the program.

The TSs change replaces the reference to Regulatory Guide (RG) 1.35 with a reference to Section XI of the ASME Code, and deletes the applicability of Surveillance Requirement 3.0.2. Compliance with RG 1.35 is not sufficient to comply with 10 CFR 50.55a, as amended, and inspection frequencies will be in accordance with Subsection IWL of Section XI; therefore, Surveillance Requirement 3.0.2 will no longer apply.

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 Containment Building is a passive safety structure that prevents the release of radioactive materials to the environment in post-accident conditions. The proposed Technical Specification change updates requirements of the Technical Specifications that have been made obsolete by the improvements of the Containment [B]uilding inspections required by the changes in the regulations. The improved inspections required by the American Society of Mechanical Engineers [Boiler and Pressure Vessel] Code serve to maintain containment response to accident conditions, by causing the identification and repair of defects in Containment Buildings.

Relocating existing requirements, eliminating requirements that duplicate regulations, and making administrative improvements provide Technical Specifications that are easier to use. Because existing requirements are controlled by regulation, there is no reduction in commitment and adequate control is still maintained. Therefore, the proposed change would not involve a significant increase in probability or consequences of an accident previously evaluated.

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

The Containment Building is a passive safety structure designed to contain radioactive materials released from the reactor coolant system. The performance of the Containment Building is not evaluated as the causal factor in any accident at Calvert Cliffs Nuclear Power Plant. The proposed Technical Specification change updates requirements of the Technical Specifications that were made obsolete by the improvements of the Containment [B]uilding inspections required by the changes in the regulations. Revising the Technical Specifications, to comply with current regulations and to eliminate duplication of requirements, does not create the possibility of a new or different [kind] of accident from any previously evaluated.

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

The safety function of the Containment Building is to provide a boundary to the release of radioactive material to the environment during post-accident conditions. The change to the Technical Specifications incorporate[s] improved inspection techniques and criteria to ensure optimum containment integrity and, therefore, optimum containment response in the event of an accident resulting in a release of radioactive material from the reactor coolant system. Optimizing containment integrity will result in maintaining the margin of safety allowed by the Containment Buildings. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Date of amendments request: September 15, 2000.

Description of amendments request: The proposed amendment revises the Unit 1 and Unit 2 Technical Specification Surveillance Requirement (SR) 3.1.7.2 which verifies that each control element assembly (CEA) not fully inserted is capable of full insertion when tripped from at least the 50 percent withdrawn position. Specifically, the proposed amendment adds a note to SR 3.1.7.2, which allows the SR to not be performed during initial power escalation following a refueling outage if SR 3.1.4.6 (CEA drop time test) has been met. In addition, ''once'' was added to the SR frequency as an administrative change to clarify that the SR is only performed once and not on a periodic basis. This proposed license amendment is consistent with Technical Specification Task Force (TSTF)–134, Revision 1, which received Nuclear Regulatory Commission (NRC) approval on April 21, 1998.

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.

A risk assessment was performed to support a prior license amendment request submitted to change Surveillance Requirement (SR) 3.1.7.2 frequency from 24 hours to 7 days. Results of a study performed in support of the risk assessment indicated no change in the geometry of those components utilized in control element assembly (CEA) insertion over the 7-day period. The study also evaluated electronic/electrical failures that could cause a CEA to be stuck, concluding that the feature that controls the movement of the CEAs is not time-related. Since there have been no modifications performed on the components analyzed or changes in the manner in which they are operated, it is reasonable to assume that the conclusions remain valid.

The CEA drop time test SR 3.1.4.6 proves that any work done during the refueling outage does not prevent the rods from tripping. Revising SR 3.1.7.2, such that it could allow more than seven days from successfully performing the CEA drop time test does not change this. However, as with any component, there will eventually be some time-related degradation that may impact the ability of the CEAs to drop. Thus, when the seven days are exceeded, there is some negligible increase in the probability that a rod would fail to drop. This causes an insignificant increase in core damage frequency because it requires multiple rod failures to cause core damage in the event of an overcooling event (the most bounding accident for a stuck CEA during rod worth testing). This additional risk is believed to be small since the degradation is the result of core changes, which occur slowly, and not the result of maintenance. Thus the risk increase due to this Technical Specification change is considered to be negligible. The probability of an overcooling event is not changed by the proposed change.

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

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

The proposed change to the surveillance requirement for CEA trippability does not result in any change to the facility or the manner in which it is operated.

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

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

Operation of the facility in accordance with this proposed amendment does not involve a significant reduction in a margin of safety. Control element assembly trippability is still demonstrated via performance of SR 3.1.4.6. The risk increase due to this change is considered to be negligible. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to provide reasonable assurance that the equipment can perform its assumed safety function.

Furthermore, this change is consistent with Technical Specification Task Force (TSTF)–134, Revision 1, which has been approved by the Nuclear Regulatory Commission. Adopting testing practices consistent with those specified in TSTF–134, Revision 1 are acceptable based on similar design, like-component testing for the system application and the availability of other Technical Specification requirements which provide regular checks to ensure limits are met.

Therefore, this proposed modification does not significantly reduce the margin of safety.

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

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

Date of amendment request: August 1, 2000.

Description of amendment request: The proposed amendments would provide revised spent fuel pool configurations, revised spent fuel pool storage criteria, and revised fuel enrichment and burnup requirements which take credit for soluble boron in maintaining acceptable margins of subcriticality in the spent fuel storage pools. Also, the proposed amendments would provide additional criteria for ensuring acceptable levels of subcriticality in the spent fuel storage pools.

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

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

No, based upon the following:

Dropped Fuel Assembly

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pools when considering the degradation of the or Boraflex panels in the spent fuel pool racks coupled with the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water, and the quantity of Boraflex remaining in the racks has no affect on the probability of such a drop accident.

The criticality analysis showed that the consequences of a fuel assembly drop accident in the spent fuel pools are not affected when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron.

Fuel Misloading

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel pool racks when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for criticality control. Fuel assembly placement and storage will continue to be controlled pursuant to approved fuel handling procedures to ensure compliance with the Technical Specification requirements. These procedures will be revised as needed to comply with the revised requirements which would be imposed by the proposed Technical Specification changes. Note that the proposed amendment would increase the number of different storage limits in Technical Specification 3.7.15. However, these revised storage limits were developed with input from station personnel. Their awareness, in conjunction with any procedure changes as described above, will provide additional assurance that an accidental misloading of a spent fuel assembly will not occur.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate soluble boron concentration. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools. A McGuire Station UFSAR change will revise Chapter 16, ''Selected Licensee Commitments'', to provide for adequate monitoring of the remaining Boraflex in the spent fuel pool racks. If that monitoring identifies further reductions in the Boraflex panels which would not support the conclusions of the McGuire Criticality Analysis, then the McGuire TS's and design bases would be revised as needed to ensure that acceptable subcriticality are maintained in the McGuire spent fuel storage pools.

Significant Change in Spent Fuel Pool Temperature

There is no significant increase in the probability of either the loss of normal cooling to the spent fuel pool water or a decrease in pool water temperature from a large emergency makeup when considering the degradation of the Boraflex in the spent fuel pool racks and the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density that would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks. However, since a reduction in the amount of Boraflex present in the racks is considered, and the spent fuel pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1470 ppm), will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling event. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a loss of normal cooling to the spent fuel pool will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

A decrease in pool water temperature from a large emergency makeup causes an increase in water density that would result in an increase in reactivity when Boraflex neutron absorber panels are present in the racks. However, the additional negative reactivity provided by the current boron concentration limit, above that provided by the concentration required to maintain k_{eff} less than or equal to 0.95 (1470 ppm), will compensate for the increased reactivity which could result from a decrease in spent fuel pool water temperature. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a decrease in pool water temperature will not be increased. Current Technical Specification 3.7.14 will ensure that an adequate spent fuel pool boron concentration is maintained in the McGuire spent fuel storage pools.

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

No. Criticality accidents in the spent fuel pool are not new or different types of accidents. They have been analyzed in Section 9.1.2.3 of the Updated Final Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 4.75 weight percent U–235. Specific accidents considered and evaluated include fuel assembly drop, accidental misloading of spent fuel assemblies into the spent fuel pool racks, and significant changes in spent fuel pool water temperature. The accident analysis in the Updated Final Safety Analysis Report remains bounding.

The possibility for creating a new or different kind of accident is not credible. The amendment proposes to take credit for the soluble boron in the spent fuel pool water for reactivity control in the spent fuel pool while maintaining the necessary margin of safety. Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility, however, a criticality accident resulting from a dilution accident was not considered credible. For the proposed amendment, the spent fuel pool dilution evaluation (Attachment 7) demonstrates that a dilution of the boron concentration in the spent fuel pool water which could increase the rack k_{eff} to greater than 0.95 (constituting a reduction of the required margin to criticality) is not a credible event. The requirement to maintain boron concentration in the spent fuel pool water for reactivity control will have no effect on normal pool operations and maintenance. There are no changes in equipment design or in plant configuration. This new requirement will not result in the installation of any new equipment or modification of any existing equipment. Therefore, the proposed amendment will not result in the possibility of a new or different kind of accident.

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

No. The proposed Technical Specification changes and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Attachment 6) based on the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' described in Reference 1. The Westinghouse methodology for taking credit for soluble boron in the spent fuel pool has been reviewed and approved by the NRC (Reference 6). This methodology takes partial credit for soluble boron in the spent fuel pool and requires conformance with the following NRC Acceptance criteria for preventing criticality outside the reactor:

(1) k_{eff} shall be less than 1.0 if fully flooded with unborated water which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

(2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The criticality analysis utilized credit for soluble boron to ensure k_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 k_{eff} calculation to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron. The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a k_{eff} of 0.95 has been evaluated (Attachment 7) and shown to be not credible. Accordingly, the required margin to criticality is not reduced.

The evaluations in Attachment 7, which show that the dilution of the spent fuel pool boron concentration from the conservative assumed initial boron concentration (2475 ppm) to the minimum boron concentration required to maintain $k_{eff} \leq 0.95$ (730 ppm) is not credible, combined with the 95/95 calculation which shows that the spent fuel rack k_{eff} will remain less than 1.0 when flooded with unborated water, provide a level of safety comparable to the conservative criticality analysis methodology required by References 2, 3 and 4.

Therefore the proposed changes in this license amendment will not result in a significant reduction in the facility's margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: September 26, 2000.

Description of amendment requests: The proposed amendments would revise the current licensing basis in the Updated Final Safety Analysis Report by requiring operator action to mitigate the effects of a loss of seal injection (LOSI) cooling to the reactor coolant pumps (RCPs).

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 of occurrence or consequences of an accident previously evaluated?

The proposed change to the licensing basis recognizes that if RCP Number 1 seal leak-off rates are low, continuous RCP operation following a sustained LOSI may no longer be permitted. Tripping the plant, securing the affected RCPs, and maintaining hot standby conditions following a sustained LOSI will permit adequate RCP seal cooling by readily achievable process controls. These actions ensure that the probability of developing excessive seal leakage equivalent to that of a previously evaluated loss of coolant accident (LOCA), has not been significantly increased. Plant and RCP tripping are anticipated transients that do not involve plant operation outside design limits.

The consequences of large- and small-break (SB) LOCAs have been evaluated and it has been shown that the radiological consequences of these events do not result in unacceptable exposures to members of the public. Therefore, even if stopping of the RCPs following a LOSI and control of process parameters as described above does not preclude RCP seal failures, the consequences of such failure are bounded by the current accident analysis.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The leakage resulting from failed RCP seals may be large enough to be considered a SBLOCA and industry data on SBLOCA initiating frequencies includes the contribution from failed RCP seals. SBLOCAs are a previously evaluated class of accidents. There is no new or different kind of accident created as a result of this change.

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

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

The original design objective for the controlled leakage seal assemblies in the RCPs was to permit sufficient controlled leakage following a LOSI, such that cooling of the leakage provided by the thermal barrier heat exchanger would be sufficient to continue RCP operation unabated without challenging seal integrity. This is an implied margin of safety for seal integrity, even if not explicitly defined in the basis for any Technical Specification. It has been postulated that the reduced seal leak-off will no longer permit continuous RCP operation following a LOSI. The proposed change to the licensing basis recognizes this condition and requires pump tripping if seal injection cannot be restored prior to receiving high temperature alarms in the leak-off return lines. Pump tripping reduces the heat generated in the pump and permits readily achievable process controls to maintain adequate seal cooling and an adequate margin to seal failure.

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

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: September 30, 2000.

Description of amendment request: The proposed amendment would allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification (T/S) Surveillance Requirement 4.4.5.3. The proposed amendment would prevent a mid-cycle shutdown to meet the required 40-calendar month inspection interval of SR 4.4.5.3 and would allow the steam generator tube inspection to be performed during the refueling outage following the current operating cycle.

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 of occurrence or consequences of an accident previously evaluated?

The accident considered applicable to the proposed change is a steam generator tube rupture (SGTR). The precursors/initiators of a SGTR (degraded, defective, or leaking tubes) are not known or expected to be present in the CNP [Cook Nuclear Plant] Unit 2 steam generators. These steam generators were newly installed in 1988, and include corrosion prevention design features not included in previous generations of steam generators.

There are no active degradation mechanisms present in the Unit 2 steam generators. Any tube imperfections that may be present or that may be initiated during the current operating cycle are not expected to progress to the point of tube failure before the next refueling outage.

Considering the condition of the steam generators and the operational time between inspections, the proposed change will not significantly increase the probability of occurrence of an accident.

The proposed change will not affect the scope, methodology, acceptance limit, or corrective measures of the existing steam generator examination program.

Unit 2 recently completed an extended shutdown that effectively limited the operational time that is the basis for the surveillance frequency. When the reactor is shut down and the reactor coolant system is at a reduced temperature, the steam generators are not subject to conditions that lead to significant tube degradation. Based on power operation time, the proposed extension will not increase the operating interval between surveillances beyond that currently allowed by [the] T/S.

The steam generator tube inspection interval is not used in the SGTR accident analysis. The proposed change will, therefore, not affect the accident analysis or methodology.

The severity of an analyzed tube rupture event is not related to the time interval between inspections. The proposed change does not affect allowable leakage rates or source terms, and does not change the duration of an SGTR or the response to the event. Because the severity of an accident is not increased by the proposed change, there is no impact on offsite dose considerations.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed change will not result in a change in plant configuration or operation. Plant systems and components will not be operated in a different manner because of this change. The proposed change does not affect or create new accident initiators or precursors.

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

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

The T/S limit of one gallon per minute total steam generator tube leakage ensures the offsite dose from tube leaks is limited to a small fraction of 10 CFR 100 limits. The T/S leakage limit of 500 gallons per day in one steam generator is based on ensuring tube integrity in the event of a steam line rupture or loss of coolant accident. Because the offsite dose considerations from steam generator tube failures are limited by the primary-to-secondary leak rate program and not the tube inspection program, the proposed change has no impact on offsite dose.

There are no tubes in service in any of the Unit 2 steam generators that were found to be degraded, and no active steam generator tube degradation is known to be occurring. Therefore, the available margin in tube wall thickness is not being significantly reduced. During the last inspection, 50% of the tubes were inspected (more than sixteen times the T/S requirement), and none were found to exceed the plugging limit, providing additional assurance that safety margins are not being reduced. The absence of tube degradation, along with the material and design features and chemistry controls, provide reasonable assurance that tube repair limits will not be approached during the current operating cycle.

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

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: June 30, 2000, as supplemented September 22, 2000.

Description of amendment request: The proposed changes would modify Sections 2.4.13.5, ''Design Bases for Subsurface Hydrostatic Loading'' 2.5.4.6.1, ''Design Basis for Groundwater'' 3.4.1.2, ''Permanent Dewatering System'' 3.8.1.6.4, ''Waterproofing Membrane'' 3.8.1.6.5, ''Steel Liner and Penetrations'' 9.3.3.1, ''Reactor Plant Vent and Drain Systems, Design Bases'' 9.3.3.2.4, ''Reactor Plant Aerated Drains System'' 9.3.3.2.4.1, ''Safety-Related Containment Recirculation Cubicle Sump'' 9.3.3.3, ''Safety Evaluation'' 9.3.3.4, ''Tests and Inspections'' and 12.3.1.3.2, ''Post-Accident Access to Vital Areas'' Tables 1.8–1, 3.2–1, 8.3–3, 12.3–3, and 12.3–4; and Figures 3.8–67 and 9.3–6 of the Final Safety Analysis Report (FSAR) to reflect the addition of the new subsystem and its impact on other safety-related systems. The new sump pump system creates the possibility of a malfunction of a different type than previously evaluated in the FSAR because of the system's dependence on electrical power; only one non-environmentally qualified, non-safety-related pump is provided; and portions of the Engineered Safety Feature Building structure are now credited with preventing Recirculation Spray System (RSS) cubicle flooding. Additionally, the proposed changes involve deviations from safety classification and ''code & standards,'' Standard Review Plan 3.4.1 and Regulatory Guide (RG) 1.26.

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

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

This license amendment request deals with changes in Millstone Unit No. 3 Final Safety Analysis Report (FSAR) due to the installation of a new sump pump system in the Engineered Safety Features Building (ESFB). The sump pump system which prevents inleakage through the containment basemat is not connected to and is fully independent of the reactor coolant system. Therefore, the proposed changes to this system will not increase the probability of occurrence of a Loss of Coolant Accident (LOCA). The new system is a support system for the Recirculation Spray System (RSS) and containment protective boundary which are mitigation design features. Therefore, the new system does not increase the probability of occurrence of accidents previously evaluated.

The proposed changes to the groundwater sump system separate the sump from the RSS pump cubicle. As such, the proposed changes would preclude flooding of the RSS cubicles and a potential malfunction of the RSS pumps. The RSS pumps function to provide containment and core cooling, as early as 11 minutes and 30 minutes, respectively, post LOCA. Operability of the RSS pumps is required long term. Since the changes do not affect the operation of the RSS pumps, they will not increase the consequences of a LOCA.

The new collection tank 3SRW–TK1 will be installed in the location of the existing abandoned in place Chemical Addition Tank (CAT) 3QSS*TK2, by the Refueling Water Storage Tank (RWST). The tank will be seismically supported utilizing similar struts and attachments to the RWST as the removed CAT. A calculation has confirmed that there is no impact on the seismic qualification of the RWST as a result of the new tank. The RWST provides water to the Emergency Core Cooling System (ECCS) and Containment Quench Spray (QSS) which are credited to mitigate the consequences of a LOCA. Therefore, the proposed changes do not increase the consequences of a LOCA.

In the proposed design, the installation of the new safety related collection sump and casing pipe will result in a change in the Supplemental Leak Collection and Release System (SLCRS) boundary within the ESFB. This modification will be performed to meet the SLCRS design requirements. Testing will be performed post modification and routinely to satisfy SLCRS Technical Specification 3/4.6.6 requirements. Per Technical Specification 3/4.6.6 basis, the SLCRS is credited post LOCA to limit the release of fission products from the containment. Since the proposed changes do not affect operability of SLCRS, it does not increase the consequences of a LOCA.

In the proposed changes, sumps 3DAS*SUMP7A/B inflow pathways will be restored such that it may become potentially contaminated. Emergency operating procedures shall contain operator actions to ensure that power to 3DAS*SUMP7A/B sump pumps 3DAS–P8A/B is isolated post LOCA. As such, the proposed changes will continue to ensure that potentially contaminated water is not discharged from 3DAS*SUMP7A/B. Therefore, the changes will not increase the consequences of a LOCA.

The design change, per NUREG–0737 Section II.B.2 requirement, has been evaluated by a calculation to ensure that the required operator actions post LOCA can be performed within a 5 rem whole body dose requirement, and has been found to be acceptable.

Therefore, these changes will not significantly increase the consequences of an accident previously evaluated.

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

This license amendment request is associated with the installation of a new sump pump system in the ESFB. The current and new sump pump systems are not accident initiators since neither system is connected to, and both are fully independent of any system that could cause an accident to occur. The new system, which collects groundwater from beneath the Containment Structure and ESFB, is a support system for the RSS and the containment protective boundary, which are design basis accident mitigation design features. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The Millstone Unit No. 3 FSAR changes reflect the installation of a new sump pump system in the ESFB. The proposed changes do not affect operation of the RWST, ECCS, QSS, RSS, SLCRS, Containment, EDG or any Class 1E component required for safety. The additional load on the Train A EDG and fuel oil consumption are within the calculated allowance. Therefore, these changes do not significantly reduce the margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: July 28, 2000.

Description of amendment request: The proposed amendment would revise the Fort Calhoun Station Unit 1 Technical Specifications to allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes.

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

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

The CE Leak Tight Sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applicable design criteria for the sleeves conform to the stress limits and margins of safety of Section III of the ASME code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits.

These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

Evaluation of the repaired steam generator tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor coolant system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Fort Calhoun Station. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The installation of the proposed sleeves is controlled via the sleeving vendor's proprietary processes and equipment. The CE process has been in use since 1984 and has been implemented more than 24 times for the installation of over 4,200 sleeves. The FCS steam generator design was reviewed and found to be compatible with the installation processes and equipment.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeved tube is bounded by the current steam generator tube rupture analysis described in Fort Calhoun Station's USAR, Section 14.14. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis, depending on the break location, and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feed line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the Fort Calhoun Station safety analysis.

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

As discussed above, the CE Leak Tight Sleeves are designed using the applicable ASME Code as guidance; therefore, they meet the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeves. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. Therefore, Omaha Public Power District concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The repair of degraded steam generator tubes with CE Leak Tight Sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Code used in the original steam generator design. The portions of the installed sleeve assembly that represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation. Use of the previously identified design criteria and design verification testing assures that the margin of safety is not significantly different from the original steam generator tubes. Therefore, OPPD concludes that the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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: July 31, 2000.

Description of amendment request: The proposed changes will revise LGS Technical Specifications (TSs) to replace the existing Automatic Depressurization System (ADS) TS Surveillance Requirement (SR) 4.5.1.d.1, a 31-day channel functional test of the accumulator backup compressed gas system low pressure alarm system, with a 31-day verification of the ADS accumulator gas supply header pressure. The existing TS SR 4.5.1.d.1 and SR 4.5.1.d.2.c, a 24-month channel calibration of the accumulator backup compressed gas system low pressure alarm system, will be relocated to the Technical Requirements Manual (TRM).

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 proposed TS changes have no physical impact on plant equipment or the normal operation of plant systems. The ADS and the ADS accumulator backup compressed gas system affected by the proposed testing changes are normally in a standby mode and there are no existing credible system failures that are accident initiators. The ability of the ADS to depressurize the vessel following a small break Loss of Coolant Accident (LOCA) so that flow from low pressure Emergency Core Cooling Systems (ECCS) can enter the core in time to limit fuel cladding temperatures is maintained by the operability of the ADS accumulators and their inlet check valves. The ADS accumulator backup compressed gas low pressure alarm system has no impact on the ability of the ADS accumulators and associated check valves to maintain an adequate gas supply required to mitigate an accident. Therefore, the removal of the alarm system testing from the TS has no impact on the ability of the ADS to cope with the small break LOCA as previously evaluated. The replacement of the monthly alarm channel functional test with the monthly verification of the ADS accumulator gas supply header pressure will assure that the ADS accumulators are pressurized as required to support ADS operability and the ability of ADS to mitigate the accident as previously analyzed is maintained. 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 have no physical impact on plant equipment or the normal operation of plant systems. The changes are limited to changes in administrative testing requirements for the existing ADS and ADS accumulator backup compressed gas low pressure alarm systems, and the long term gas supply to the ADS valves. The changes do not impact the methods of operation or manipulation of these systems or components. The impact of these changes has been evaluated to assure that the changes are in conformance with the required design and licensing basis, and that system performance is not degraded. The changes do not affect the operation of the ADS or the ADS accumulator backup gas system and do not create any new system failure modes or accident initiators not previously considered. 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 changes maintain the safety design basis of the ADS and the ADS accumulator backup gas systems. The ADS accumulator backup compressed gas low pressure alarm system does not support the operability of the ADS accumulators which are required to maintain an adequate gas supply for ADS vessel depressurization. Therefore, the Channel Functional Test and Channel Calibration of backup gas system alarms can be removed from the TS and have no impact on the ability of the ADS to depressurize the reactor and maintain current safety margins defined in the design basis for this TS. The availability of the ADS accumulator backup gas system to perform its long term cooling function after an accident or other event is not addressed in any TS or Bases. The proposed changes in testing also do not impact any of the Inservice Inspections or Tests currently performed on the ADS or ADS accumulator backup gas system components. 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.

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

Date of amendment request: February 29, 2000.

Description of amendment request: The amendment would incorporate Supplement 3 to PL–NF–90–001, ''Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements,'' into Technical Specification Section 5.6.5, Core Operating Limits Report. The supplement describes alternative methods for the analysis of the rotated bundle event, the control rod withdrawal error event, and the recirculation flow controller event.

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

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

[The] proposed alternative analysis methods do not involve an increase in the probability or consequences of an accident previously evaluated. The alternative analysis methods affect the analysis methods used to perform the Rotated Bundle Analysis, the Rod Withdrawal Error Analysis, and the Recirculation Flow Controller Failure Analysis. These events are analyzed on a cycle specific basis to ensure that the operating limits contained in the COLR [Core Operating Limits Report] will provide acceptable consequences to the health and safety of the public consistent with NRC guidelines. No physical changes are being made to plant systems, structures or components. The alternative analysis methods ensure that the [offsite] dose consequences of the postulated events remain within the NRC guidelines.

Based on the above, it is concluded that the alternative analysis methods do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The alternative analysis methods do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed alternative analysis methods affect the analysis methods for the Rotated Bundle, Rod Withdrawal Error and Recirculation Flow Controller Failure Events. Since these alternative analysis methods affect analytical methods and do not affect any plant systems, structures, or components, it is concluded that the proposed alternative analysis methods do not create the possibility for any new or different kind of accident from any accident previously evaluated.

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

The alternative analysis methods do not involve a significant reduction in the margin of safety.

The Rotated Bundle Methodology is currently analyzed as a moderate frequency event. The alternative methods will instead analyze the Rotated Bundle Event as an Infrequent Event. Analysis of this event as an infrequent event is consistent with NRC guidance (provided in the Standard Review Plan) and the frequency classification of the event as described in the SSES [Susquehanna Steam Electric Station] FSAR [Final Safety Analysis Report]. The proposed analysis methodology limits the analytical [offsite] dose to a small fraction of 10 CFR 100 guidelines consistent with the NRC guidelines. Therefore, the proposed alternative analysis methods do not represent a significant reduction in the margin of safety.

The Rod Withdrawal Error Analysis currently does not credit the Rod Block Monitor System to limit the extent of the inadvertent rod withdrawal. The alternative proposed methods will allow credit in the analysis for the Rod Block Monitor to limit the extent of the inadvertent control rod withdrawal. Several plant and procedural improvements have been implemented that have improved the reliability of the Rod Block Monitor System. The analytical acceptance criteria for the event is not affected. Therefore, the proposed alternative analysis methods do not affect the margin of safety.

The Recirculation Flow Controller Failure analysis is currently analyzed using the RETRAN code. The proposed alternative analysis methods use [PPL Susquehanna, LLC's] approved steady state nodal simulation methodology instead of the RETRAN code. The [PPL Susquehanna, LLC,] steady state nodal simulation methodology produces final operating limits that are consistent with the RETRAN methodology. The analytical acceptance criteria is not affected by the alternative analysis methodology. Use of [PPL Susquehanna, LLC's] methodology does not affect 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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: July 31, 2000.

Description of amendment request: The amendment would remove the phrase ''maximum pathway'' from Surveillance Requirement 3.6.1.3.12 in Technical Specification Section 3.6.1.3, ''Primary Containment Isolation Valves.''

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

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

The proposed change to eliminate the words ''maximum pathway'' does not affect any plant system or component. The change does not impact operator performance or procedures. The leak rate testing of the MSIVs [main steam isolation valves] will continue to be performed in accordance with

10 CFR 50 Appendix J. The change does not impact the design basis accident analyses presented in the FSAR [Final Safety Analysis Report]. The change only affects how the as-found leakage is used to evaluate operability and reportability. This change is consistent with the guidance on leak rate testing presented in NEI 94–01 [Nuclear Energy Institute Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J] and the Standard Technical Specifications. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As discussed above, the proposed change to the Technical Specifications does not affect any plant system or component and does not affect plant operation. The consequences of accidents will remain within the accident analysis described in the FSAR. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect any plant system or component, and does not have any impact on plant operation. The proposed change does not involve a significant reduction in the margin of safety as currently defined in the bases of the applicable Technical Specification section. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment requests: September 6, 2000 (PCN–274, Supplement 1). This application supersedes the licensee's application of November 24, 1999.

Description of amendment requests: The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Southern California Edison Company to withdraw its November 24, 1999, application for proposed amendments. The Commission had previously issued a Notice of Consideration of Issuance of Amendments published in the **Federal Register** on December 29, 1999 (64 FR 73098). However, by letter dated September 6, 2000, the licensee withdrew the proposed change. TAC Nos. MA7289 and MA7290 used for the review of the November 24, 1999, application have been closed.

As submitted by the licensee on September 6, 2000, the proposed amendments would modify the Technical Specifications (TSs) for the San Onofre Nuclear Generating Station, Units 2 and 3, to revise TS 3.3.11, ''Post Accident Monitoring Instrumentation (PAMI).'' Specifically, the proposed change would extend the PAMI channel calibration surveillance frequency from 18 months to 24 months to accommodate a 24-month fuel cycle for all PAMI instruments with the exception of the reactor coolant system (RCS) temperature instrumentation. Surveillance Requirement (SR) 3.3.11.4 relating to RCS temperature instrumentation channel calibration every 18 months will remain in place.

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:

Do the proposed amendments:

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

Response: No

The proposed license amendment to extend the calibration surveillance frequency of Post Accident Monitoring Instrumentation (PAMI) (excluding RCS temperature instrumentation) is being made to support plant operation with a 24-month fuel cycle. Increasing the calibration intervals for PAMI instrumentation to 30 months [24 months plus the 25 percent surveillance interval extension allowed by SR 3.0.2] (excluding RCS temperature instrumentation) does not affect the initiation or probability of any previously analyzed accident. Increasing the calibration interval will not affect the integrity of any of the principal barriers against radiation release (fuel cladding, reactor vessel, and containment building). The ability of the plant to mitigate the consequences of any previously analyzed accidents is not adversely affected.

PAMI instrumentation provides to the operators both qualitative and quantitative information used in accident mitigation and for the safe shutdown of the plant. Instrumentation which provides qualitative information is unaffected by a change in instrument accuracy induced by drift due to the increased surveillance interval because no explicit value is required by the Emergency Operating Instructions (EOIs). Instrumentation that provides quantitative information (i.e., decision points) in the EOIs have been evaluated. This evaluation resulted in no changes to any operating instructions. This evaluation of the proposed change to the surveillance interval demonstrates that licensing basis safety analyses acceptance criteria and San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 EOI criteria will continue to be met.

The proposed new surveillance frequency for these instrument channels was evaluated using the guidance of Generic Letter 91–04 [''Changes in Technical Specification Surveillance Intervals To Accommodate a 24-Month Fuel Cycle'']. The basis for the change includes a quantitative evaluation of instrument drift for PAMI instrumentation (excluding RCS temperature instrumentation) providing quantitative information to the EOIs. Also, loop accuracy/setpoint calculations for these instruments were updated to accommodate the extended surveillance period. Analyses and evaluations completed to assess the proposed increase in the surveillance interval demonstrate that the effectiveness of these instruments in fulfilling their respective functions is maintained. Technical Specifications Channel Checks and Channel Functional Checks for the subject channels, will continue to be performed to provide assurance of instrument channel OPERABILITY.

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

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

Response: No

The increased calibration surveillance interval for PAMI instrumentation (excluding RCS temperature instrumentation) is justified based on evaluation of past equipment performance and does not require any plant hardware changes or changes in normal system operation. Changing the calibration interval for this instrumentation has no means of creating the possibility of a new or different kind of accident. There are no new decision points or operator responses required to support existing accident mitigation strategies.

Therefore, there are no new failure modes introduced as a result of extending these surveillance intervals, and the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change to the calibration surveillance interval (excluding RCS temperature instrumentation) was evaluated using the criteria of 95% probability/95% confidence level for process sensor drift.

PAMI instrumentation are used to provide indication following certain hypothetical accident conditions and are used in EOIs for trending and to initiate operator action at certain decision points. Instrument uncertainty calculations have been updated for PAMI instrumentation used for EOI decision points as appropriate. Updated calculations show that the total loop uncertainty for PAMI evaluated either decreased or remained the same. These updated calculations demonstrate that applicable accuracy requirements for SONGS 2 and 3 are satisfied with the proposed new surveillance intervals.

Changing the calibration interval for these channels does not affect the margin of safety for previously analyzed accidents. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application for amendments: August 31, 2000 (TS 99–17).

Brief description of amendments: The proposed amendment would revise the Sequoyah Nuclear Plant (SQN) Technical Specifications (TSs). The revision would revise TS Section 5.6, ''Fuel Storage,'' to allow credit for soluble boron in the fuel storage pools.

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

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

The presence of soluble boron in the spent fuel pool (SFP) water for criticality control does not increase the probability of a fuel assembly misplacement accident in the SFP. The handling of the fuel assemblies in the SFP has always been performed in borated water. The proposed change does allow greater flexibility for fuel storage configurations in the SFP. The increased flexibility does not introduce any greater complexity than the 3-zone configuration now in use. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the TS limitations. There is no increase in the probability of a fuel placement accident.

The criticality analysis shows the consequences of the most serious fuel assembly misplacement accident in the SFP are not affected when considering the presence of soluble boron. Under normal conditions, the rack k_{eff} [k effective] remains subcritical as required by 10 CFR 50.68 [Section 50.68 of Title 10 of the Code of Federal Regulations], and is less than 0.95 with only 300 ppm [parts per million] soluble boron concentration. In the event of a postulated fuel assembly misplacement, the presence of sufficient soluble boron in the SFP precludes criticality as a result of the misplacement. The criticality analysis demonstrates that the pool k_{eff} will remain less than 0.95 following an accidental misplacement due to 2000 parts per million (ppm) boron concentration of the pool. In fact, concentration of only 700 ppm soluble boron is sufficient to maintain k_{eff} less than 0.95 with 95% probability at 95% confidence level for the most serious fuel assembly misplacement. The proposed TS will ensure that an adequate SFP boron concentration is maintained. There is no significant increase in the consequences of the accidental misplacement of spent fuel assemblies in the SFP.

There is no increase in the probability of the loss of normal cooling to the SFP water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

Reactivity changes due to SFP temperature changes have been evaluated. The base case criticality analysis used a SFP temperature of 20°C. The SFP reactivity uncertainty due to temperature changes was considered for SFP temperatures ranging from 4°C to 120°C. The reactivity increment between 4°C and 20°C is taken into account as additional uncertainty in the analysis. In all spent fuel temperature cases, the temperature (and void) coefficients of reactivity are negative. Therefore there is no requirement for additional soluble boron above the base case level. Because the coefficients of reactivity are negative, the consequences of the loss of normal cooling to the SFP will not be increased.

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

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

Spent fuel handling accidents are not new or different types of accidents and have been evaluated in the criticality analysis, Reference 1.

The boron concentration in the SFP water is maintained at a minimum of 2000 ppm. The proposed changes to the TS do not change boron concentration requirements for the SFP water. A dilution of the SFP soluble boron has always been a possibility; however, it was shown in the SFP dilution evaluation (Reference 2) that there are no credible dilution events for which the SFP k_{eff} could reach criticality. Therefore, the implementation of proposed changes to the TS will not result in the of a new kind of accident.

The proposed changes for re-rack storage management continue to specify requirements for the spent fuel rack configurations. Since the proposed SFP storage configuration limitations are comparable to those used in the past, the new limitations will not have any significant effect on normal SFP operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The misplacement of a fuel assembly in the revised storage configurations has been evaluated. In all cases, the rack k_{eff} remains subcritical and less than 0.95 with 700 ppm boron in the water.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident. There is no significant change in plant configuration, equipment design, or equipment.

Under the proposed amendment, no changes are being made to the racks themselves, any other systems, or to the physical fuel handling structures in the Auxiliary Building itself. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The TS changes proposed by this License Amendment Request and the resulting spent fuel storage configuration limitations will provide adequate safety margin to ensure that the storage fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Reference 1) performed in accordance with accepted spent fuel rack criticality analysis methodology.

While the criticality analysis utilized partial credit for soluble boron, storage configurations have been defined to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide subcritical margin such that the SFP k_{eff} is maintained less than 0.95 under all credible conditions.

The loss of substantial amounts of soluble boron from the SFP, which could lead to k_{eff} exceeding 0.95, has been evaluated (Reference 2) and shown to be not credible. This evaluation also shows that dilution of the SFP boron concentration from 2000 ppm to 800 ppm is not credible. Also, the spent fuel storage pool k_{eff} remains less than 1.0 at a 95/95 probability/confidence level with the pool filled with unborated water. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above evaluation, TVA concludes that the proposed changes to the TSs does [sic] not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: September 14, 2000, as supplemented on September 22, 2000.

Description of amendment request: This proposed change revises the Technical Specification to clarify the valve isolation signal information in Table 4.7.2 and makes an administrative change to the table main steam isolation valves component identification to include all the valves.

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 operation of the Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

No changes are being made to plant design, method of operation or method of testing. This change will not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed change does not affect the ability of the primary containment isolation system or ECCS [emergency core cooling system] systems to perform their required safety functions. The essential safety functions of providing primary containment integrity and providing water to cool the core in the event of an accident are maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or conclusions of existing safety analyses. This proposed change results in no impact on analyzed accident event precursors or effects.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. There is no change in plant operation that involves failure modes other than those previously evaluated. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

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

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

No changes are being made to plant design, method of operation or method of testing. This change will not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed change does not affect the ability of the primary containment isolation system or ECCS systems to perform their required safety functions. The essential safety functions of providing primary containment integrity and providing water to cool the core in the event of an accident are maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or conclusions of existing safety analyses. This proposed change results in no impact on analyzed accident event precursors or effects.

This proposed change does not alter the physical design of the plant, methods or modes of operation, testing or analyses, thereby resulting in no impact on safety functions. Since the proposed change does not alter the means by which primary containment isolation is maintained and containment cooling valves are isolated in support of RHR [residual heat removal] LPCI [low pressure coolant injection] actuation, there is no significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 19, 2000.

Description of amendment request: This proposed change revises Technical Specification (TSs) 3.5.H.3 and 3.5.H.4 related to low pressure Emergency Core Cooling System (ECCS) injection/spray subsystem operability during cold shutdown and refueling conditions. Two circumstances are considered: (1) when no operations with the potential for draining the reactor vessel (OPDRV) are in progress (addressed in TS 3.5.H.3), and (2) when OPDRVs are in progress (addressed in TS 3.5.H.4). The proposed change provides completeness in the TS for the defined conditions and also provides for the operation of an alternative combination of low pressure ECCS injection/spray subsystems to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel during cold shutdown and refueling conditions when OPDRVs are in progress.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

No changes are being made to plant design or method of operation. This change only affects the plant in a cold shutdown or refueling condition and will not alter the basic operation of process variables, structures, systems, or components as described in the safety analyses. No new equipment is introduced.

The proposed change does not affect the ability of low pressure ECCS injection/spray systems to perform their required safety functions. The essential safety function of providing water to reflood the reactor vessel following an inadvertent vessel draindown is maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or conclusions of existing safety analyses.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its intended safety function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. There is no change in plant operation that involves failure modes other than those previously evaluated. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

During refueling and cold shutdown conditions with operations having the potential for draining the reactor vessel (OPDRV) in progress, any one ECCS injection/spray subsystems is adequate to reflood the reactor vessel in the event of an inadvertent draindown. Since the proposed change provides an equivalent means for achieving this safety function, there is no reduction in reflood capability. The additional flexibility, to maintain a combination of one core spray subsystem and one LPCI [low pressure coolant injection] subsystem (provided by this change), is equivalent to the safety margin provided by the existing TS since a single active failure affecting one subsystem results in the same remaining capability of one ECCS subsystem.

Since the changed TS provides equivalent low pressure ECCS injection/spray capability and protection from loss of coolant inventory, the risk of an inadvertent draindown event is unchanged, thus preserving previously existing margins of safety.

For circumstances involving no OPDRVs during refueling and cold shutdown conditions, no ECCS or containment cooling equipment is required to meet safety objectives. Thus, the margins of safety for such situations are maintained.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 26, 2000.

Description of amendment request: This proposed change revises Technical Specification (TS) requirements regarding secondary containment systems, including the Standby Gas Treatment System (SBGTS). The affected TS sections are 1.0, Definitions; 3/4.7.B, Standby Gas Treatment System; and 3/4.7.C, Secondary Containment System. In addition, a new TS section, 3/4.7.E, Reactor Building Automatic Ventilation System Isolation Valves (RBAVSIVs), is proposed. Some of the proposed changes are administrative in nature and do not affect the technical aspects of the requirements. Associated changes to the TS Bases are also being made to conform to the changed TS. The proposed changes provide certain additional flexibility in operations when equipment is made or found to be inoperable, while also ensuring appropriate actions are taken to place the plant in a safe condition under such conditions.

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 operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

No changes are being made to the plant design, physical system configuration, or basic method of operation as a result of the proposed amendment. The Standby Gas Treatment System (SBGTS) and secondary containment are not assumed to be initiators of any analyzed event. The circumstances for which operability of SBGTS and secondary containment are required are unchanged, and would not occur at any greater frequency as a result of this change. Therefore, the probability of a design basis loss-of-coolant accident or fuel handling accident (the applicable accidents) previously evaluated is not increased.

The proposed change does not increase the consequences of an accident because system operability requirements are being maintained. In lieu of suspending refueling activities when one train of SBGTS is inoperable beyond seven days, placing the operable train of SBGTS in operation ensures that no failures that could prevent automatic actuation have occurred and that any other failure would be readily detected. Operation of one train of the SBGTS is sufficient to mitigate the consequences of any analyzed event. The secondary containment systems assumed to operate following a design basis accident continue to function as assumed in accident analyses to mitigate the consequences of postulated accidents.

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

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve any physical alteration to the plant structures, systems, or components (SSCs), or the basic manner in which these SSCs are operated or maintained. The methods by which these systems perform their safety function are unchanged and remain consistent with current safety analysis assumptions. There is no change in plant operation that involves failure modes other than those previously evaluated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed change does not result in a significant reduction in a margin of safety because restrictions placed on operations which have the potential for releasing radioactive material to the secondary containment continue to be in accordance with the assumptions and conditions of existing safety analyses. Operations with inoperable equipment have the proper restrictions to maintain existing margins or to place the plant in a safe condition such that inoperable equipment is not required to meet safety analysis assumptions. Ensuring operability of one train of SBGTS together with required secondary containment integrity is sufficient to mitigate the consequences of any analyzed event. Since current analyses are unaffected in this regard, margins of safety are maintained.

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

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

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

NRC Section Chief: James W. Clifford.

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

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: September 1, 2000.

Description of amendment request: The proposed amendment would revise certain 18-month surveillance requirements in the technical specifications by eliminating the condition that testing be conducted during shutdown, or during cold shutdown or refueling mode. The systems that would be affected are the emergency core cooling system, containment depressurization and cooling system, chemical addition system, and containment isolation valve system.

Date of publication of individual notice in **Federal Register**: September 12, 2000 (65 FR 55056).

Expiration date of individual notice: October 12, 2000.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: April 27, 2000.

Brief description of amendment request: The amendment seeks to extend the applicability of the current pressure-temperature and overpressure protection system limit curves from 13.3 effective full-power years (EFPY) to 16.2 EFPYS.

Date of publication of individual notice in ***Federal Register:*** August 29, 2000 (65 FR 52451).

Expiration date of individual notice: September 28, 2000.

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

Date of amendment request: August 29, 2000, as supplemented by letter dated September 8, 2000.

Description of amendment request: The amendment proposes to change Technical Specifications 3.0.D and 4.0.D to be equivalent to the Boiling-Water Reactor NUREG–1433 guidance for the Improved Technical Specifications limiting condition for operation 3.0.4, which is currently under review.

Date of publication of individual notice in **Federal Register:** September 14, 2000 (65 FR 55650).

Expiration date of individual notice: October 16, 2000.

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

Date of amendment request: September 14, 2000.

Brief description of amendment request: The amendment would clarify the valve isolation signal information in the Technical Specification Table 4.7.2 and make an administrative change to the table main steam isolation valves component identification.

Date of publication of individual notice in ***Federal Register:*** September 27, 2000 (65 FR 68111).

Expiration date of individual notice: October 27, 2000.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible 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. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: June 19, 2000, as supplemented August 8, 2000.

Brief description of amendment: The amendment allows some emergency diesel generator Technical Specification surveillance requirements to be performed during plant operation instead of during plant shutdown.

Date of issuance: October 2, 2000.

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

Amendment No.: 132.

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

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46006). The supplemental information did not change the application or affect the proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 1, 1999, as supplemented June 14, and July 27, 2000.

Brief description of amendment: The amendment revised the TMI–1 Technical Specifications (TSs) 1.4.2, 1.4.3, 1.4.4, 3.3.1.2.b, 3.3.1.3.b, and c, 3.3.2.1, Table 4.1–1 (Items 14, 25, 31, and 32), Table 4.1–3 (Items 4 and 6), Table 4.1–5, and TSs 4.1.5, 4.5.2.1.a and b, 4.5.2.3.a, and 4.5.3.1.b.1 and 2, to: add limiting condition for operation (LCO) action statements and make LCOs and surveillance requirements more consistent with the revised ''Standard Technical Specifications for Babcock & Wilcox Plants,'' (NUREG–1430, Revision 1); correct conflicts or inconsistencies; and revise spent fuel pool sampling frequency from monthly and after adding chemicals, to weekly. TS 3.3.1.2.d is deleted as a result of the LCO additions described above. Also, a Bases statement for surveillance testing was added to Section 4.1 of the TSs and a revised Bases to Section 4.4.4 is included as well.

Date of issuance: September 25, 2000.

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

Amendment No.: 225.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 28, 1999 (64 FR 40906) and August 23, 2000 (65 FR 51349).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 25, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 1, 1999 (102–04378).

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

Date of issuance: September 29, 2000.

Effective date: September 29, 2000, to be implemented within 30 days of the date of issuance.

Amendment Nos.: Unit 1—128, Unit 2—128, Unit 3—128.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating Licenses.

*Date of initial notice in **Federal Register:*** March 8, 2000 (65 FR 12288).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 6, 2000, as supplemented June 29 and July 3, 2000.

Brief description of amendments: The amendments restrict the emergency diesel generator (DG) acceptance criteria for steady-state voltage and frequency in several surveillance requirements (SRs) involving DG starts in Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' of the TSs for the three units. The amendments also add a note to each SR that states ''The steady state voltage and frequency limits are analyzed values and have not been adjusted for instrument error.'' The restricted acceptance criterion is to ensure proper DG operation.

Date of issuance: October 4, 2000.

Effective date: October 4, 2000, to be implemented within 45 days of the date of issuance. For surveillance requirements associated with the revised steady-state voltage and frequency limits in Technical Specifications 3.8.1 and 3.8.2, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of the amendments.

Amendment Nos.: Unit 1–129, Unit 2–129, Unit 3—129.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 12, 2000 (65 FR 43043).

The June 29 and July 3, 2000, supplements provided clarifying information that was within the scope of the application and the **Federal Register** notice, and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 18, 2000.

Brief description of amendments: The amendments remove the anticipatory reactor scram signal for turbine electro-hydraulic control (EHC) low oil pressure trip from the reactor protection system (RPS) trip function.

Date of issuance: September 27, 2000.

Effective date: Immediately, to be implemented within 90 days.

Amendment Nos.: 181 and 176.

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

*Date of initial notice in **Federal Register:*** April 5, 2000 (65 FR 17910).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 27, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: April 25, 2000.

Brief description of amendments: The amendments revised Technical Specification 3/4.9.5, ''Communications'' to allow the movement of a control rod in a fueled core cell in Operational Condition 5 to be exempt from the requirement that direct communication be maintained between the control room and the refueling platform personnel when the rod is moved with its normal drive system.

Date of issuance: October 5, 2000.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 141 and 127.

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

*Date of initial notice in **Federal Register:*** June 14, 2000 (65 FR 37422).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 5, 2000.

No significant hazards consideration comments received: No.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: June 27, 2000, as supplemented August 18 and 30, 2000.

Brief description of amendment: The amendment changes Improved Technical Specification Sections 3.5.1, ''Safety Injection Tanks (SITs),'' and 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' regarding completion times for restoring an inoperable SIT and for restoring a low-pressure safety injection train.

Date of issuance: October 2, 2000.

Effective date: As of the date of issuance and shall be implemented on or before December 31, 2000.

Amendment No.: 191.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46007) (two notices).

The August 18 and 30, 2000, supplemental letters provided clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 29, 2000, as supplemented by letters dated July 27, and August 10, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to reference the Westinghouse Best Estimate Large Break Loss-of-Coolant Accident analysis methodology described in WCAP–12945–P–A, March 1998. These amendments also address corresponding TS Bases changes.

Date of issuance: October 2, 2000.

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

Amendment Nos.: 188 and 181.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51350).

The letter dated August 10, 2000, provided additional information that did not change the scope of the application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 25, 2000, as supplemented by letters dated July 31, August 8, and August 17, 2000.

Brief description of amendments: The amendments temporarily revise TS 3.5.2, ''Emergency Core Cooling System;'' TS 3.6.6, ''Containment Spray System;'' TS 3.6.17, ''Containment Valve Injection Water System;'' TS 3.7.5, ''Auxiliary Feedwater System;'' TS 3.7.7, ''Component Cooling Water System;'' TS 3.7.8, ''Nuclear Service Water System;'' TS 3.7.10, ''Control Room Area Ventilation System;'' TS 3.7.12, ''Auxiliary Building Filtered Ventilation Exhaust System;'' and TS 3.8.1, ''AC Sources''.

Date of issuance: October 4, 2000.

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

Amendment Nos.: 189 and 182.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 25, 2000 (65 FR 51860).

The supplements dated July 31, August 8, and August 17, 2000, provided clarifying information that did not change the scope of the May 25, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

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

Date of application of amendments: September 7, 2000.

Brief description of amendments: The amendments revise Surveillance Requirement 3.8.1.9.a by adding a note stating that the upper limits on frequency and voltage are not required to be met for the annual test of the Keowee Hydro Units until the NRC issues an amendment that removes the note in response to an amendment request to be submitted no later than April 5, 2001.

Date of Issuance: October 4, 2000.

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

Amendment Nos.: 316, 316, & 316.

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

Public comments requested as to proposed no significant hazards consideration: Yes (65 FR 56600 dated September 19, 2000). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 19, 2000, but indicated that if the Commission makes a final no significant hazards consideration.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated October 4, 2000.

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: May 8, 2000, as supplemented by letter dated August 30, 2000.

Brief description of amendment: The amendment revises Technical Specifications to remove the fuel building (FB) and the FB ventilation system from the requirements associated with secondary containment during power operation (except during movement of recently irradiated fuel assemblies in the FB).

Date of issuance: September 22, 2000.

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

Amendment No.: 113.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37424).

The August 30, 2000, supplemental letter provided additional information to support staff review of the original application, and did not affect the initial finding of no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO–1 and ANO–2), Pope County, Arkansas

Date of application for amendments: September 17, 1999, as supplemented by letters dated June 29, August 3, and September 15, 2000.

Brief description of amendment: The amendments change heavy load handling requirements and transportation provisions that would permit the movement of the original and replacement steam generators (SGs) through the ANO–2 containment construction opening during the SG replacement outage.

Date of issuance: September 25, 2000.

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

Amendment Nos.: 209 & 221.

Facility Operating License Nos. DRP–51 and NPF–6: The amendments revise the licenses.

Date of initial notice in ***Federal Register:*** February 23, 2000 (65 FR 9004).

The additional information provided in the June 29 and August 3, 2000, supplemental letters was noticed in the **Federal Register** on August 23, 2000 (65 FR 51352). The September 15, 2000, supplement provided clarifying information that was within the scope of the **Federal Register** notice published August 23, 2000, and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 25, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 29, 1999, as supplemented by letters dated January 26, May 17 (2 letters), May 31, and August 4, 2000.

Brief description of amendment: The amendment revised the License and Technical Specifications (TSs), and corresponding Bases have been changed to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators (SGs) that are being used for Cycle 15 operation. The License was revised to incorporate a new methodology employed in calculating radiological doses for some non-loss-of-coolant accident events. TS changes were made to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) low pressurizer pressure setpoints, the RPS and ESFAS low SG pressure setpoints, the RPS and ESFAS low SG level setpoints, the reactor coolant flow rate limit, and the high linear power trip setpoints with inoperable main steam safety valves (MSSVs). The amendment also made changes to the TSs and corresponding Bases have been changed that are not directly related to the replacement SGs. These changes revised the allowed outage time of the MSSVs in Modes 1 and 2 to allow up to 12 hours to reduce the high linear power level-high trip setpoint when one or more MSSVs are inoperable, and revised the action statement in Mode 3 to maintain at least two MSSVs operable on each SG.

Date of issuance: September 29, 2000.

Effective date: As of the date of issuance to be implemented prior to startup from the 2R14 refueling outage.

Amendment No.: 222.

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

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6405).

The January 26, May 17 (2 letters), May 31, and August 4, 2000, supplemental letters 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 consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 29, 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: January 12, 2000, as supplemented by letters dated June 15, 2000, and September 7, 2000.

Brief description of amendment: The proposed changes modify Technical Specification (TS) 3.9.4, ''Containment Building Penetrations,'' to allow the containment equipment door, airlocks, and other penetrations to remain open, but capable of being closed, during core alterations or movement of irradiated fuel in containment. Additionally, a note, Bases changes, and Surveillance Requirements changes provide further enhancements to clarify equipment door, airlock, and penetration closure capability.

Date of issuance: October 2, 2000.

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

Amendment No.: 169.

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

Date of initial notice in ***Federal Register:*** February 23, 2000 (65 FR 9008). The June 15, 2000, and September 7, 2000, supplemental letters provided clarifying information that did not expand the scope of the original **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 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: September 20, 1999, as supplemented May 12, 2000.

Brief description of amendments: The amendments revised the standard to which the control room ventilation charcoal and Supplementary Leak Collection and Release System (SLCRS) charcoal must be laboratory tested as specified in: BVPS–1 Technical Specification (TS) 4.7.7.1.1.c.2 for the Control Room Emergency Habitability Systems; BVPS–1 TS 4.7.8.1.b.3 for the SLCRS; BVPS–2 TS 4.7.7.1.d for the Control Room Emergency Air Cleanup and Pressurization System; and BVPS–2 TS 4.7.8.1.b.3 for the SLCRS. Nuclear Regulatory Commission Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999, requested licensees to revise their TS criteria associated with laboratory testing of ventilation charcoal to a valid test protocol, which included American Society for Testing and Materials (ASTM) D3803–1989. These license amendments revised the charcoal laboratory standard to follow ASTM D3803–1989 for each BVPS Unit. These license amendments also: (1) Revised the minimum amount of output in kilowatts needed for the control room emergency ventilation system heaters at each BVPS unit; (2) revised BVPS–1 SLCRS surveillance testing criteria to be consistent with American Nuclear Standards Institute/American Society of Mechanical Engineers N510–1980, the BVPS–1 control room ventilation testing, and BVPS–2 SLCRS/control room ventilation testing; and (3) made minor typographical corrections.

Date of issuance: September 29, 2000.

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

Amendment Nos.: 234 and 117.

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

Date of initial notice in ***Federal Register:*** August 29, 2000 (65 FR 52449).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 1, 2000, as supplemented July 21, 2000.

Brief description of amendment: The amendment: (1) Revised Technical Specification (TS) requirements regarding the minimum number of radiation monitoring instrumentation channels required to be operable during movement of fuel within the containment; (2) revised the Modes in which the surveillance specified by Table 4.3–3, ''Radiation Monitoring Instrumentation Surveillance Requirements,'' Item 2.c.ii is required; (3) revised TS 3.9.4, ''Containment Building Penetrations,'' to allow both personnel air lock (PAL) doors and other containment penetrations to be open during movement of fuel assemblies within containment, provided certain conditions are met; (4) revised applicability and action statement requirements of TS 3.9.4. to be for only during movement of fuel assemblies within containment; (5) revised periodicity and applicability of Surveillance Requirement (SR) 4.9.4.1; (6) revised SR 4.9.4.2 to verify flow rate of air to the supplemental leak collection and release system (SLCRS) rather than verifying the flow rate through the system; (7) added two new SRs, 4.9.4.3 and 4.9.4.4, for verification and demonstration of SLCRS operability; (8) modified TS 3/4.9.9 for the containment purge exhaust and isolation system to be applicable only during movement of fuel assemblies within containment; (9) revised associated TS Bases and made editorial and format changes; and, (10) revised the BVPS–2 Updated Final Safety Analysis Report (UFSAR) description of a fuel-handling accident (FHA) and its radiological consequences. The changes to the BVPS–2 UFSAR reflect a revised FHA analysis that the licensee performed to evaluate the potential consequences of having containment penetrations and/or the PAL open during movement of fuel assemblies within containment. These UFSAR revisions include potential exclusion area boundary, low population zone, and control room operator doses as a result of an FHA.

Date of issuance: September 28, 2000.

Effective date: As of date of issuance. Technical Specification changes shall be implemented within 60 days.

Amendment No: 116.

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

Date of initial notice in **Federal Register:** August 23, 2000 (65 FR 51342).

The July 21, 2000, letter provided clarifying information that did not expand the scope of the amendment and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 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: February 16, 2000.

Brief description of amendments: (1) Accident monitoring instrumentation for both St. Lucie Units 1 and 2, (2) motor operated valve thermal overload protection bypass device TS for Unit 2, and (3) an administrative change to the Unit 2 Technical Specification (TS) Index.

Date of Issuance: October 4, 2000.

Effective Date: October 4, 2000.

Amendment Nos.: 165 and 109.

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

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17916).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: November 22, 1999, as supplemented August 14, 2000.

Brief description of amendment: The amendment would adopt selected NRC approved generic changes to the Improved Technical Specifications (ITS) NUREGs. The 16 changes come from the Technical Specification Task Force (TSTF) process developed by the Industry and the NRC. Three of these changes are Bases-only changes but are included for completeness relative to the TSTF process.

Date of issuance: October 3, 2000.

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

Amendment No.: 234.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 12, 2000 (65 FR 1924).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: November 29, 1999, as supplemented by letter dated May 2, 2000.

Brief description of amendment: The amendment changes Technical Specification (TS) 3/4.6.6, ''Supplementary Leak Collection and Release System''; TS 3/4.7.7, ''Control Room Emergency Ventilation System''; TS 3/4.7.9, ''Auxiliary Building Filter System''; and TS 3/4.9.12, ''Fuel Building Exhaust System''; in response to Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: October 4, 2000.

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

Amendment No.: 184.

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

Date of initial notice in **Federal Register:** January 26, 2000 (65 FR 4287).

The letter dated May 2, 2000, provided clarifying information and did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the application as published in the **Federal Register**.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 18, 2000.

Brief description of amendment: The amendment changes the Technical Specifications to add operability requirements for the No. 12 residual heat removal service water pump.

Date of issuance: October 2, 2000.

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

Amendment No.: 113.

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

Date of initial notice in **Federal Register:** August 23, 2000 (65 FR 51361).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, PSEG Nuclear LLC, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: June 14, 2000, as supplemented August 9, 2000.

Brief description of amendment: This amendment revised the TSs for safety limit Minimum Critical Power Ratio from its current value of 1.10 to 1.09 for two recirculation-loop operation, and from 1.12 to 1.10 for single recirculation-loop operation.

Date of issuance: September 22, 2000.

Effective date: As of date of issuance, and shall be implemented prior to startup for Cycle 14 operations, scheduled for October 2000.

Amendment No.: 236.

Facility Operating License No. DPR–44: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46012). The August 9, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: April 27, 2000.

Brief description of amendment: The amendment would extend the applicability of the current pressure-temperature limit curves and overpressure protective system setpoints from 13.3 to 16.2 effective full-power years.

Date of issuance: October 5, 2000.

Effective date: October 5, 2000.

Amendment No.: 202.

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

Date of initial notice in ***Federal Register:*** August 29, 2000 (65 FR 52431).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 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: August 29, 2000, as supplemented September 8, 2000.

Brief description of amendment: The amendment adapts the provisions of the Boiling Water Reactor Standard Technical Specifications (STS) regarding applicability of Technical Specifications 3.0.D and 4.0.D in the event of plant shutdown.

Date of issuance: September 29, 2000.

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

Amendment No.: 262.

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

Public comments requested as to proposed no significant hazards consideration: Yes September 14, 2000 (65 FR 55650). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 16, 2000, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment finding of exigent circumstances, state consultation, and final determination of no significant hazards consideration determination are considered in a Safety Evaluation dated September 29, 2000.

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: April 27, 2000, as supplemented September 5, 2000.

Brief description of amendment: The amendment changes the Trip Level Settings for the Residual Heat Removal and Core Spray Start Timers as well as the Automatic Depressurization System Auto-Blowdown Timer.

Date of issuance: October 4, 2000.

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

Amendment No.: 263.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37428).

The September 5, 2000, supplement did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 2, 2000.

Brief description of amendments: The amendments modify the requirements in Technical Specifications Section 3/4.6.3, ''Containment Isolation Valves,'' by changing limiting conditions for operation (LCO) 3.6.3.1 and 3.6.3 for Unit Nos. 1 and 2, respectively. The changes delete the asterisk (*) modifying the word OPERABLE in LCO 3.6.3.1 (Unit 1) and LCO 3.6.3 (Unit 2), and relocate its associated footnote to the Action portion of the LCO.

Date of issuance: October 2, 2000.

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

Amendment Nos.: 235 and 216.

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

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 39959). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 13, 2000

Brief description of amendments: The amendments revise TS Table 3.3–6, ''Radiation Monitoring Instrumentation,'' by changing the Containment Gaseous Activity Monitor (R12A) alarm and trip setpoint for the containment purge and pressure relief system isolation for Mode 6 (Refueling) operations.

Date of issuance: October 2, 2000.

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

Amendment Nos.: 236 and 217.

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

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46013).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 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: May 3, 2000 (PCN–516), as supplemented August 25, 2000.

Brief description of amendments: The amendments consist of changes to the Technical Specifications that revise the pressure temperature (P–T) limits for 20 effective full power years and reduce the minimum boltup temperature from 86 °F to 65 °F. The P–T limits calculations are based on the 1989 American Society of Mechanical Engineers Appendix G methodology.

Date of issuance: September 28, 2000.

Effective date: September 28, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—172; Unit 3—163.

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

*Date of initial notice in **Federal Register:*** May 31, 2000 (65 FR 34749). The supplemental letter dated August 25, 2000, provided clarifying information that was within the scope of the May 3, 2000, application and the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: November 17, 1999, as supplemented by letter dated August 21, 2000.

Brief description of amendments: The amendments revise TS 5.5.7, ''Ventilation Filter Testing Program'' to include the requirements for laboratory testing of Engineered Safety Feature Ventilation System charcoal samples in accordance with American Society Testing and Materials D3803–1989 and the application of a safety factor of 2.0 to the charcoal filter efficiency assumed in the plant design-basis dose analyses. In addition, editorial revisions are being made to some portions of TS Section 5.0 to reference the correct sections of Regulatory Guide 1.52.

Date of issuance: October 3, 2000.

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

Amendment Nos.: 223 and 164.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 15, 1999 (64 FR 70091). The supplemental letter dated August 21, 2000, provided clarifying information that did not change the scope of the November 17, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 3, 2000.

No significant hazards consideration comments received: Yes. One comment was received, and is addressed in the above-referenced Safety Evaluation.

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

Date of amendment request: September 28, 1998, as supplemented on April 22, 1999, April 27, 2000, and August 15, 2000.

Brief description of amendments: The amendments revise the technical specifications (TSs) to eliminate the need to enter TS 3.0.3 when multiple trains of either the control room makeup and cleanup filtration system or the fuel handling building exhaust air system are inoperable by providing an allowed outage time of up to 12 hours to restore at least one train to an operable status.

Date of issuance: September 26, 2000.

Effective date: September 26, 2000, to be implemented within 60 days.

Amendment Nos.: Unit 1—125; Unit 2—113.

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

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46016). The August 15, 2000, submittal provided clarifying information that was within the scope of the revised application and **Federal Register** notice and did not change the staff's revised proposed no significant hazards considerations determination issued on July 26, 2000. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 26, 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: May 16, 2000.

Brief description of amendments: These amendments change the Technical Specifications (TSs) by replacing Surveillance Requirement (SR) 4.8.1.1.2.c, for evaluating fuel oil for the emergency diesel generators, with a Diesel Fuel Oil Program in Section 6. The revision also deletes the portion of the SRs that specifies the use of sodium hypochlorite solution in cleaning of the fuel oil storage tanks, deletes the SR to perform a pressure test on the diesel generator fuel oil system designed to American Society of Mechanical Engineers Section III requirements, and corrects various typographical errors in the TS and Bases. Two Bases pages are also added to each units TS. The applicable TS Bases are also revised.

Date of issuance: October 2, 2000.

Effective date: October 2, 2000.

Amendment Nos.: 261 and 252.

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

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48758). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 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: February 4, 2000.

Brief description of amendments: These amendments change the Technical Specifications (TS) to revise the cold leg accumulator volume and pressure limits based on instrumentation changes, instrument inaccuracies, and instrumentation tap locations. The applicable TS bases are also revised.

Date of issuance: October 6, 2000.

Effective date: October 6, 2000.

Amendment Nos.: 262 and 253.

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

*Date of initial notice in **Federal Register:*** May 17, 2000 (65 FR 31360). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 25, 2000 (ULNRC–04258).

Brief description of amendment: The amendment expands (1) The range of acceptable lift settings for the pressurizer safety valves (PSVs), and (2) the tolerance (from ±1% to ±2%) of the as-found, measured lift settings of tested PSVs, to be operable. Following testing, however, the lift settings of the PSVs would remain nor more than the current ±1%. The amendment revises Technical Specifications (TS) 3.3.2, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' 3.4.10, ''Pressurizer Safety Valves,'' and 3.4.11, ''Pressurizer Power Operated Relief Valves (PORVs),'' of the Callaway TS. For TS 3.3.2, a new Action H for one or more trains inoperable is added, the note for surveillance requirement (SR) 3.3.2.14 is revised to identify another slave relay that the SR would be applicable to, and the automatic PORV actuation is added to Table 3.3.2–1, ''Engineered Safety Features Actuation System Instrumentation.'' For TS 3.4.10, the range of allowable PSV lift settings in the limiting condition for operation (LCO) is expanded from ≥2460 and ≤2510 to ≥2411 and ≤2509, and SR 3.4.10.1 is revised to state that, following testing, the lift settings shall be ''within 1% of 2460 psig'' instead of simply ''within 1%.'' The nominal PSV lift setting would be changed from 2485 psig to 2460 psig. For TS 3.4.11, Actions A and B is revised to be actions for inoperable PORVs either solely due to excessive PORV seat leakage (Action A) or for reasons other than excessive seat leakage (Action B), and Action E would remain an action for two inoperable PORVs, but would be only for reasons other than excessive seat leakage.

Date of issuance: September 25, 2000.

Effective date: September 25, 2000, to be implemented (including issuing the revised EOP E–O and training all the control room operator crews on the revised procedure) before the restart from refueling outage 11, the next refueling outage for Callaway Plant, Unit 1, scheduled to begin in Spring 2001.

Amendment No.: 137.

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

Date of initial notice in ***Federal Register:*** June 28, 2000 (65 FR 29964). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 25, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 21, 2000 (ULNRC–04285), as supplemented August 16, 2000.

Brief description of amendment: The amendment revises Limiting Condition for Operation (LCO) 3.9.4, ''Containment Penetrations,'' of the Callaway Technical Specifications (TS) to allow containment penetrations with direct access to the outside atmosphere to be open under administrative controls during refueling operations, by adding a note to the LCO that states ''containment penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.'' In addition, there is a format and editorial correction to TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Start Air,'' to correct an error in the conversion to the improved TS issued May 28, 1999, in Amendment No. 133.

Date of issuance: September 26, 2000.

Effective date: September 26, 2000, to be implemented (including the completion of the administrative procedures that ensure that open containment penetrations, with direct acess to the outside atmosphere during refueling operations with core alterations and irradiated fuel movement inside containment, will be promptly closed in the event of a fuel handling accident inside containment) before refueling operations during refueling outage 11.

Amendment No.: 138.

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

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51364). The August 16, 2000, supplement provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 26, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 8, 1999.

Brief description of amendment: The amendment authorizes revisions to the descriptions of the steam generator tube rupture and main steam line break accidents in the Callaway Plant, Unit 1 Final Safety Analysis Report (FSAR) to reflect increases in the radiological dose consequences calculated by the licensee for these accidents.

Date of issuance: September 27, 2000.

Effective date: September 27, 2000, to be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the FSAR the changes to the description of the facility as described in the licensee's application dated September 8, 1999, and evaluated in the staff's Safety Evaluation attached to the amendment.

Amendment No.: 139.

Facility Operating License No. NPF–30: The amendment revised the Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** October 6, 1999 (64 FR 54383). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 11th day of October 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–26645 Filed 10–17–00; 8:45 am]

BILLING CODE 7590–01–P

PRESIDIO TRUST

Presidio Theatre (Building 99), The Presidio of San Francisco, California, Notice of Termination of Environmental Impact Statement Process

AGENCY: The Presidio Trust.

ACTION: Notice of termination of Environmental Impact Statement (EIS) process for the rehabilitation and expansion of the Presidio Theatre (Building 99) within The Presidio of San Francisco, San Francisco, California (Presidio).

RATIONALE: The Presidio Trust (Trust) is terminating the EIS process for the Presidio Theatre in order to complete an update of the 1994 General Management