the working papers at the Website, select ''Final Rule on Radiological Criteria for License Termination,'' then select ''Lic Term Document Library,'' then select ''Regulatory Guide,'' and then select ''Module C.2: Regulatory Position—Final Status Survey,'' or ''Module C.1: Regulatory Position—Dose Modeling.''

Meeting Agenda

9:00 Welcome and introduction

9:05 Presentation describing issues considered in developing the draft working paper

10:30 Break

10:45 Public comments on the draft working paper. Attendees will be asked for questions and comments on each section of the draft working paper.

12:00 Lunch

1:30 Continuation of public comments.

5:00 Adjourn

Submitting Written Comments

Comments may be posted electronically on the NRC Technical Conference Forum Website mentioned above. Comments submitted electronically can also be viewed at that Website. Comments may also be mailed to the Chief, Rules and Directives Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

FOR FURTHER INFORMATION: For information or questions on meeting arrangements, contact Nina Barnett, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone 301–415–6187, fax 301–415–5385, E-mail: NMB@NRC.GOV. For technical information or questions, contact Stephen A. McGuire, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone 301–415–6204; fax: 301–415–5385; E-mail: SAM2@NRC.GOV.

Dated at Rockville, Maryland this 22nd day of January, 1998.

For the Nuclear Regulatory Commission.

Cheryl Trottier,

Chief, Radiation Protection and Health Effects Branch, Division of Regulatory Applications, RES.

[FR Doc. 98–2017 Filed 1–27–98; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 5, 1998, through January 15, 1998. The last biweekly notice was published on January 14, 1998 (63 FR 2271).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By February 27, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

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

Date of amendment request: December 17, 1997.

Description of amendment request: The requested amendment revises Technical Specification Section 5.6.5, ''Core Operating Limits Report (COLR).'' The revisions add reference to an additional approved methodology for correlating departure from nucleate boiling (DNB) ratios. The added methodology is the Siemens Power Corporation Topical Report, EMF–92–153(P)(A), ''HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel.''

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

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

The proposed change adds a methodology that has been previously reviewed and approved by the NRC for determining the DNB safety limit. The new methodology utilizes the High Thermal Performance (HTP) correlation developed by the fuel manufacturer, Siemens Power Corporation. The HTP correlation is empirically based and results in a DNB safety limit that corresponds to a 95% probability at a 95% confidence level that DNB will not occur. The DNB ratio safety limit is a conservative design value which is used as a basis for setting core safety limits. The DNB correlation is not assumed to be an initiator of analyzed events or transients, and use of the new DNB correlation will not alter assumptions relative to mitigation of accident or transient events. The proposed change has been confirmed to ensure that no previously evaluated accident or transient results in a DNB less than the DNB correlation safety limit. The HTP DNB correlation assures with high confidence that, for accidents and transients that do not result in a DNBR less than the HTP DNBR safety limit, departure from nucleate boiling and subsequent fuel overheat will not occur in HTP fuel.

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

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

The proposed change does not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed change will allow use of the new DNB correlation in like manner as the existing DNB correlation in the analysis of accidents and transients to assure that the acceptance criteria for current analyses are met. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows use of a DNB correlation that determines a safety limit that is slightly lower than the currently used DNB correlation. While the slightly lower DNB correlation safety limit allows a small increase in margin in analyzing accidents and transients, the change from the existing DNB correlation to the proposed DNB correlation is not directly comparable to the margin of safety. This is because the margin of safety for a particular accident or transient is that margin that results from the difference between the DNBR calculated for the particular accident or transient using the DNB correlation and the DNBR safety limit determined by the DNB correlation. Since both the safety limit and the accident or transient calculated DNB use the same DNB correlation, the margin of safety is consistently calculated and evaluated for acceptability. Since both the current and proposed DNB correlation closely approximate test data, and they still meet the 95/95 criterion, and the new DNB correlation does not result in a DNBR from an accident or transient less than the DNBR correlation safety limit, the proposed change does not result in a significant reduction in the margin of safety.

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

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

NRC Project Director: Gordon E. Edison, Acting.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: June 6, 1997, as supplemented September 25, 1997.

Description of amendment request: The proposed amendment would delete the requirement to sample the spray additive tank per Technical Specification (TS) Table 4.1–2, ''Frequency for Sampling Tests,'' and delete the sodium hydroxide (NaOH) reference in TS Section 5.2.C.1. The request to delete the requirement and the reference was inadvertently omitted as part of the licensee's original submittal dated August 22, 1996, supplemented March 28, 1997, to eliminate the requirement for the NaOH containment spray additive and spray additive tank.

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

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

Response:

The request to remove the requirement for the spray additive tank was approved as part of Amendment No. 191 to Operating License No. DPR 26. By letter dated April 23, 1997, the Commission reviewed and approved the amendment request. However, Consolidated Edison failed to include the deletion of the requirement to sample the spray additive tank. The removal of the requirement for the spray additive tank has been analyzed and approved; therefore, there is no further basis for continued testing of the tank. Further, the deletion of the requirement would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response:

The proposed changes allow the containment safeguards to mitigate the consequences of a design basis LOCA [loss-of-coolant accident] in a manner equivalent to that previously approved. Therefore, the proposed changes do not create an accident or malfunction of safety equipment of a different type.

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

Response:

With the proposed changes, all of the safety criteria previously evaluated are still valid and remain conservative. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: December 17, 1997.

Description of amendment request: The proposed amendments would revise Section 6.9.1.9 of the Technical Specifications (TS) to reference updated or recently approved topical reports, which contain methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report. These topical reports have all been previously approved by the staff under licensing actions separate from the current amendment request.

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

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

No. The proposed changes do not involve any modification to existing systems, components, operating limits, or operating procedure. Therefore, these proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

No. No actual plant equipment or operating procedure will be affected by the proposed changes. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

No. Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design or operation. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28242–0001.

NRC Project Director: Herbert N. Berkow.

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: December 17, 1997.

Description of amendment request: The proposed amendments would revise Section 6.9.1.9 of the Technical Specifications (TS) to reference updated or recently approved topical reports, which contain methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report. These topical reports have all been previously approved by the staff under licensing actions separate from the current amendment request.

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

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

No. The proposed changes do not involve any modification to existing systems, components, operating limits, or operating procedure. Therefore, these proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

No. No actual plant equipment or operating procedure will be affected by the proposed changes. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

No. Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design or operation. Thus, the margin of safety previously analyzed and evaluated is maintained.

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.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina.

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

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

Date of amendment request: September 23, 1997.

Description of amendment request: The proposed amendment changes the Reactor Protective System and Engineering Safety Actuation System trip set point and allowable values for steam generator low pressure. The proposed amendment also relocates the RPS and ESFAS response time tables from the Technical Specifications to the Safety Analysis Report.

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 Not Involve a Significant Increase in the Probability or Consequences of and Accident Previously Evaluated.

The proposed changes included in this amendment request do not affect the accident initiators in any of the accidents previously evaluated. The proposed trip setpoints and allowable values for Steam Generator Pressure—Low are being reduced by this proposed amendment request. This change is necessary to increase the operating margin between the full power steam generator pressure and these setpoints. The change should reduce the probability of an inadvertent Main Steam Isolation Signal (MSIS) from occurring at power since it will increase the operating space between the operating pressure and the setpoints. Therefore, this amendment request will not increase the probability of any accident previously evaluated.

The secondary system pipe break safety analyses were reanalyzed for the Steam Generator Pressure—Low setpoint reduction effort. This effort included the removal of unnecessary analysis conservatisms resulting in a significant reduction in the associated setpoints. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment used to mitigate the consequences of an accident. The previously evaluated accidents which were determined to be impacted by this setpoint change were evaluated with no significant increase in the consequences.

This amendment request contains the relocation of the Reactor Protective System (RPS) and Engineered Safety Features Actuation System (ESFAS) response time information from the Technical Specifications (TS) to the Safety Analysis Report. This proposed change adopts the TS ''line-item improvement'' as recommended in NRC Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' dated December 29, 1993. The NRC has concluded that 10 CFR 50.36 does not require the response time tables to be retained in TSs and has issued Generic Letter 93–08 as a line item improvement to allow their removal. Response time testing will still be required by the ANO–2 TS after the relocation of the associated response time information in this amendment request. Relocating the response time information for the RPS and ESFAS from the TS to the SAR will not alter these surveillance requirements. Therefore, the relocated response time portion of this amendment request is considered administrative in nature and will not affect the probability or consequences of any accident previously evaluated.

Therefore, this change does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does Not Create the Possibility of a New or Different Kind of Accident from and Previously Evaluated.

The proposed changes do not involve any physical modifications (i.e., new systems, new components, etc.) to the plant. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment used to mitigate the consequences of an accident. The results of the accident reanalyzes suggest no different phenomena or plant behavior than previously considered. The Steam Generator Pressure Low setpoint change does not create any new or different system actuations or interactions than evaluated previously. The relocated response time portion of this amendment request is considered administrative in nature and is not considered an accident initiator. Therefore, this change does *not* create the possibility of a new or different kind of accident from any previously evaluated.

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

The accidents which were determined to be impacted by the Steam Generator Pressure Low setpoint change were evaluated to ensure acceptable results are maintained. The instrument error calculations supporting the lower Steam Generator Pressure Low setpoint and allowable values will ensure the present accident analysis assumptions are still maintained. The methodology used to determine the instrument loop errors and uncertainties is the same as that used in previous amendment requests that have been reviewed and approved by the NRC. Based on these evaluations, the proposed changes do *not* involve a significant reduction in a margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the reauested chance does not involve a significant hazards consideration.

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John Hannon.

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

Date of amendment request: September 23, 1997.

Description of amendment request: The proposed amendment reduces the minimum primary system flow that is specified in the technical specifications to reflect the effects of increased primary system resistance caused by steam generator tube plugging.

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 Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

Entergy Operations is proposing a change to the Technical Specifications for Arkansas Nuclear One—Unit 2 (ANO–2) to accommodate a larger number of plugged steam generator tubes. The proposed amendment request will revise the Technical Specifications to conservatively account for the reduced reactor coolant system (RCS) flow effects of plugging up to 30 percent of the tubes in either steam generator. This change will reduce the minimum RCS total flow rate from 120.4×10^6 lbm/hr to 108.4×10^6 lbm/hr until the steam generators are replaced. The steam generators are currently scheduled for replacement during the fall of the year 2000. After the steam generators are replaced, the minimum RCS flow will then return to the current value of 120.4×10^6 lbm/hr.

The tube plugs that are installed in the steam generators are passive components by nature. This amendment request does not change the type of plugs which may be installed in the steam generators nor does it change the criteria for plugging steam generator tubes. Reducing the minimum required RCS flow does not change the plant's required mode of operation or modify any active component. Therefore, this amendment request will not significantly increase the probability of the occurrence of a previously evaluated accident.

The installation of steam generator tube plugs removes the affected tube from service thus reducing the heat transfer surface area and increasing the steam generator primary side flow resistance. The increased flow resistance in the affected steam generator leads to a reduction in the RCS flow available for core cooling. The reduced RCS flow rate and heat transfer surface area resulted in a change in several primary and secondary parameters that required reanalysis. The ANO–2 accident reanalyses supporting the additional steam generator tube plugging and the reduction in RCS flow have been completed.

The Design Basis Accidents (DBAs) affected by these changes were reanalyzed to determine if the effects of increased steam generator tube plugging and the reduced RCS flow could result in exceeding the acceptance criteria applicable to each of these events. It was determined that the DBA acceptance criteria would *not* be exceeded as a result of increased steam generator tube plugging and reduction in the minimum RCS flow rate.

Based on the results of the analysis, it is concluded that the emergency core cooling system design satisfies the acceptance criteria of 10 CFR 50.46(b) for a spectrum of small break and large break loss of coolant accidents (LOCAs). The specified acceptable fuel design limits (SAFDLs) and the RCS pressure boundary limits also are not violated. The fuel and core performance were also determined to remain within acceptable limits. Primary and secondary system pressures remain below their respective pressure limits.

Analyses and evaluations of the DBAs have been performed demonstrating that the NRC acceptance criteria for these events are met. The revised analyses and evaluations consider reduced RCS flow, increased RCS temperatures, and increased steam generator tube plugging conditions. Although the offsite dose during a steam generator tube rupture event could increase, the results remain well within 10 CFR [Part] 100 limits. Therefore, the consequences of a previously evaluated accident are not significantly increased.

Therefore, this change does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed amendment reduces the minimum RCS total flow to account for the effects of steam generator tube plugging. This amendment request will not change the modes of operation defined in the Technical Specifications. This change does not add any new equipment, modify any interfaces with any existing equipment, change the equipment's function, or the method of operating the equipment. The proposed change does not change plant conditions in a manner which could affect other plant components. Reactor core, RCS, and steam generator parameters remain within appropriate design limits during normal operation. The proposed change could not cause any existing equipment to become an accident initiator. Therefore, this change does *not* create the possibility of a new or different kind of accident from any previously evaluated.

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

The margins of safety associated with this change are defined in the fuel and core related analyses, and in each of the transient and accident analyses affected by the reduced RCS flow. An evaluation of the affected analyses confirmed that the established acceptance criteria for specified acceptable fuel design limits, primary and secondary system over-pressurization, and the acceptance criteria for the emergency core cooling systems have been satisfied by this license amendment request. The evaluation concludes that, when considering the proposed Limiting Conditions for Operation for the minimum RCS total flow rate, all applicable acceptance criteria limits are met. The margins of safety associated with the transient and accident analyses affected by this change will not be significantly reduced. Therefore, this change does *not* involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does *not* involve a significant hazards consideration.

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: John Hannon.

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

Date of amendment request: December 10, 1997.

Description of amendment request: To clarify certain sections of the Technical Specifications (TSs) and Bases which have been demonstrated to be unclear or conflicting. Administrative changes include TS 2.3 Bases, Table 3.1.1.G.1, Table 3.1.1.M.2, Section 4.3.C, and Section 6.1.1. Technical changes include Table 3.3.3, note b, Section 3.4 Bases, Section 3.8 Bases and Section 4.5 Bases.

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:

With respect to the administrative changes, they are typical of the example I.c.2.e.i in 51 FR 7744 and therefore, they do 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 the margin of safety; in that they are purely administrative changes to achieve consistency or correct an error in the TS.

With respect to technical change, Table 3.1.1, note b:

1. Involve a siginificant increase in the probability or consequences of an accident previously evaluated; (or)

The proposed change would restore the original value of less than 600 psig. This lower value would not increase the probability of any accident as it provides a more conservative level below which protection can be bypassed.

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

The proposed change would restore the original value of less than 600 psig. The setpoint of a bypass cannot create a different kind of accident, it can only affect the severity.

3. Involve a significant reduction in a margin of safety; As the requested change lowers the bypass setpoint, the margin of safety will be increased.

With respect to Section 3.4 Bases:

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

The proposed change to the Bases removes a possible area of confusion from the [TS], and updates the Bases to reflect the results of newer, approved methodologies. Therefore, no change to any probability calculation occurs.

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

The proposed change addresses an existing accident (Small Break LOCA) and removes outdated and possibly confusing information. Therefore, no new or different kind of accident is created.

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

The proposed change does not change the way the plant is operated or the way design Bases are maintained. It only removes an outdated and possibly confusing paragraph from the Bases, therefore, no margin of safety is affected.

With respect to Section 3.8 Bases:

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

The Isolation Condenser Radiation Monitors had no impact o[n] the operation of any plant system. Additionally, the monitors were not relied upon for any post accident evaluations. They were removed from the plant using the 10 CFR 50.59 process. As this request updates the [TS] Bases to reflect the plant as currently configured, no impact on the probability or consequences of any previously evaluated accident is possible.

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

The Isolation Condenser Radiation Monitors had no impact o[n] the operation of any plant system. Additionally, the monitors were not relied upon for any post accident evaluations. They were removed from the plant using the 10 CFR 50.59 process. As this request updates the [TS] Bases to reflect the plant as currently configured, no new or different kind of accident is created.

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

The Isolation Condenser Radiation Monitors had no impact o[n] the operation of any plant system. Additionally, the monitors were not relied upon for any post accident evaluations. They were removed from the plant using the 10 CFR 50.59 process. As this request updates the [TS] Bases to reflect the plant as currently configured, no reduction in any margin of safety can occur.

With respect to Section 4.5 Bases:

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

No change to any procedure, nor any modification to any system is requested. The same surveillance will be performed at the same frequency. Only the brand of chemical used to perform the surveillance will be affected. As an equivalent chemical will be selected, no increase in the probability or consequences of an accident previously evaluated can be created.

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

No change to any procedure, nor any modification to any system is requested. The same surveillance will be performed at the same frequency. Only the brand of chemical used to perform the surveillance will be affected. As an equivalent chemical will be selected, no new or different kind of accident previously evaluated can be created.

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

No change to any procedure, nor any modification to any system is requested. The same surveillance will be performed at the same frequency. Only the brand of chemical used to perform the surveillance will be affected. As an equivalent chemical will be selected, no margin of safety can be affected.

The 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.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for licensee: Ernest L. Blake, Jr., Esquire. Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Ronald B. Eaton.

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

Date of amendment requests: October 3, 1997.

Description of amendment requests: The proposed amendment would revise the Operating License to allow the start of core offload as soon as 60 hours after shutdown instead of the 120 hours currently specified.

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 Operating License Amendment will not significantly increase the probability or consequences of any previously evaluated accidents.

The proposed change will allow initiation of core offload earlier after shutdown than is currently allowed. Thermal-hydraulic analysis shows that maximum bulk SFP, local water, and fuel clad temperatures will remain within acceptable limits and, in fact, do not exceed those previously reviewed and approved for Amendment 195.

Thermal-hydraulic analysis shows the minimum time to action is calculated at 4.5 hours versus 5.5 hours previously reviewed and approved for Amendment 195. In the event of a loss of forced cooling with cask pit isolation gate failure event, the DAEC will use Emergency Service Water (ESW), a Seismic Category I system, to provide makeup to the SFP. It is estimated to take no more than 2 hours to provide ESW makeup to the SFP, therefore the minimum time to action of 4.5 hours is sufficient time to prevent uncovering the fuel in the SFP.

The DAEC design basis refueling accident, as discussed in Section 15.10.2 of the Updated Final Safety Analysis Report, assumes a twenty-four hour decay time before core offload begins. The proposed change does not adversely affect that accident analysis.

Therefore, the proposed change will not result in an increase in probability or consequences of an accident previously evaluated.

2. The proposed changes will not create a new or different kind of accident from those previously evaluated.

Thermal-hydraulic analysis shows that the proposed change will not result in maximum bulk SFP, local water, or fuel clad temperatures which would initiate bulk pool boiling, challenge fuel rod integrity or jeopardize the structural integrity of the pool.

As stated above, the minimum time to action of 4.5 hours allows sufficient time to provide ESW makeup to the SFP. Therefore, this change does not create the possibility of a new or different type of accident.

3. The proposed change will not result in a significant reduction in any margin of safety.

This change will not result in maximum bulk SFP, local water, and fuel clad temperatures in excess of those previously evaluated and accepted per Amendment 195. The thermal-hydraulic analysis for Case C does show a reduction in the minimum time to action by one hour. However, 4.5 hours does provide sufficient time to provide ESW makeup to the SFP as this task is estimated to require no more than 2 hours. Furthermore, this change does not result in any change to the Technical Specifications. Therefore, this change does not result in a significant reduction in a margin of safety.

Based upon the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

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

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, Iowa 52401.

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

Date of amendment request: December 15, 1997.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 2.1 and 3/4.4.1 to change the safety limit minimum critical power ratio (MCPR) for the upcoming fuel operating cycle (Cycle 7) from 1.07 to 1.09 for two recirculation loop operation and from 1.08 to 1.10 for single loop operation. An obsolete footnote in TS 3/4.4.1, which states that ''the MCPR Safety Limit of 1.07 will be used through the first operating cycle,'' would be deleted. The associated Bases 2.1 would be changed to (1) reflect the new MCPR values, (2) delete certain details (including Bases Table B2.1.2–1, ''Uncertainties Used in the Determination of the Fuel Cladding Safety Limit,'' and Bases Table B2.1.2–2, ''Nominal Values of Parameters Used in the Statistical Analysis of Fuel Cladding Integrity Safety Limit,'') and (3) substitute for the deleted detail a reference to General Electric Standard Application for Reactor Fuel (GESTAR II), NEDE–24011–P–A, and to the cycle-specific analysis. The TS Index would be changed to reflect deletion of Bases Tables B2.1.2–1 and B2.1.2–2.

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 Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the revised Safety Limit MCPR was performed using the NRC approved methodology in GESTAR II. The Safety Limit MCPR is a TS numerical value that cannot initiate an event. Maintaining compliance with this limit will assure that 99.9 percent of the fuel rods will not experience transition boiling during transient events. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. The proposed change does not modify any of the accident initiators described in the USAR [Updated Safety Analysis Report]. No equipment malfunctions or procedural errors are created as a result of this change, therefore, no accidents are affected by it. The change does not adversely impact the integrity of the fuel cladding, which is the first barrier to the release of radioactivity to the environment. The change does not affect the operation of any systems necessary to mitigate the radiological consequences of an accident or to safely shutdown the plant. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, 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 Safety Limit MCPR is a TS numerical value designed to prevent fuel damage from transition boiling. It cannot create the possibility of a transient or accident. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. The proposed change does not directly impact the operation of any systems or equipment important to safety. The analyses show that all fuel licensing acceptance criteria are met. The fuel cladding, reactor vessel, and reactor coolant system integrity will be maintained. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The Safety Limit MCPR calculation was performed using the NRC approved methodology in GESTAR II. Analyses of limiting USAR transients establish Operating Limit MCPR values that ensure that the Safety Limit MCPR is not violated. The revised cycle specific Safety Limit MCPR preserves the existing margin of safety and will continue to assure that 99.9 percent of the fuel rods will not experience transition boiling during transient events. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. Thus, the margin of safety to fuel cladding failure due to insufficient cladding heat transfer during transient events is not reduced. Therefore, this 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: S. Singh Bajwa.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: November 13, 1997.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to (1) modify the low temperature overpressure protection (LTOP) requirements; (2) modify the reactor coolant system (RCS) heatup and cooldown limits; and (3) make changes to correct various items based on the licensee's review of the current TSs. The supporting TS Bases sections would also be changed to reflect the proposed TS changes.

The affected TSs are: TS 3.1.2.1, ''Flow Paths—Shutdown;'' TS 3.1.2.2, ''Flow Paths—Operating;'' TS 3.1.2.3, ''Charging Pump—Shutdown;'' TS 3.1.2.4, ''Charging Pumps—Operating;'' TS 3.1.2.5, ''Boric Acid Pumps—Shutdown;'' TS 3.1.2.6, ''Boric Acid Pumps—Operating;'' TS 3.1.2.8, ''Borated Water Sources—Operating;'' TS 3.4.1.3, ''Coolant Loops and Coolant Circulation—Shutdown;'' TS 3.4.3, ''Relief Valves;'' TS 3.4.9.1, ''Reactor Coolant System;'' TS 3.4.9.2, ''Pressurizer;'' TS 3.4.9.3, ''Overpressure Protection Systems;'' TS 3.5.3, ''ECCS Subsystems—T_{avg} < 300 °F;'' and TS 3.10.3, ''Pressure/Temperature Limitation—Reactor Criticality.''

The November 13, 1997, submittal provides specific details related to each of the proposed changes.

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.

Each of the proposed changes have been grouped together, as appropriate, to address this criteria.

HPSI Pump Not Required To Be Operable In Modes 5 and 6. The proposed change to only require one charging pump to be operable in Modes 5 and 6, instead of the current requirement for one charging pump and one high pressure safety injection pump (HPSI) pump to be operable, will result in sufficient, but not excessive, Reactor Coolant System (RCS) makeup capability. When the plant is in Mode 5 or 6 there are two major factors to consider with respect to the number of RCS makeup pumps required to be operable. If too many RCS makeup pumps are required, an inadvertent start of these pumps can result in a mass addition transient beyond the capacity of the Low Temperature Overpressure Protection (LTOP) System. This may result in an RCS pressure increase that exceeds the 10CFR50 Appendix G pressure/temperature limits. Compliance with the mass input and venting restrictions contained in the proposed Technical Specification 3.4.9.3 will ensure the Appendix G limits are not exceeded.

The minimum number of RCS makeup pumps required to be operable in Modes 5 and 6 ensures sufficient makeup capability is available for RCS inventory control and RCS boration requirements. RCS inventory control is necessary in Modes 5 and 6 to ensure sufficient water is available for core cooling. A rapid loss of RCS inventory due to catastrophic pipe failures is unlikely in Modes 5 and 6 due to the reduced RCS pressure and temperature. An inventory loss is more likely to occur due to small system component failures or during infrequently performed evolutions, such as reduced inventory operation. This type of inventory loss will occur at a slower rate. Plant operators will have time to perform the necessary actions to mitigate the event. Reliance on automatic operation of the Emergency Core Cooling System is not necessary and Technical Specifications do not require automatic actuation by the Engineered Safety Features Actuation System to be operable in Mode 4 or below. Operator action is sufficient to mitigate a loss of RCS inventory in Mode 4 or below, provided sufficient

RCS makeup capability is available. Plant procedures and shutdown risk management will provide adequate administrative control to ensure sufficient RCS makeup capability is available, or that contingency plans have been developed.

The minimum number of RCS makeup pumps required to be operable in Modes 5 and 6 ensures sufficient makeup capability is available for RCS boration requirements. The RCS is required to be borated to a sufficient boron concentration to ensure the Technical Specification Shutdown Margin (SDM) requirements are met. The appropriate SDM requirements must be met before entry is allowed into Mode 5 or 6. RCS boron concentration is increased to establish the required SDM. This is normally accomplished by adding borated water to the RCS during plant cooldown to compensate for the contraction of the RCS inventory. The proposed change will restrict the number of pumps available, which will increase the time required to adequately borate the RCS. However, the change will not affect the ability to add boric acid to the RCS.

Even though the proposed change will remove the Technical Specification requirement for an operable HPSI pump in Modes 5 and 6, sufficient RCS makeup capability will be available to meet RCS boration and inventory requirements. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

LTOP Mass Input and Vent Size Requirements. The proposed changes to the RCS venting requirements currently contained in Technical Specification 3.4.9.3, and the RCS makeup requirements that will be relocated to Technical Specification 3.4.9.3 are necessary to be consistent with the new LTOP analysis. These changes will ensure the 10CFR50 Appendix G limits are not exceeded.

The proposed changes to the mass input restrictions will still allow two charging pumps and one HPSI pump to be capable of injecting into the RCS when the RCS is operating in Mode 4 [less than or equal to] 275 °F. This combination will be allowed in Mode 5 until RCS temperature is [less than or equal to] 190 °F. When RCS cold leg temperature is at or below 190 °F only one charging pump will be allowed to be capable of injecting into the RCS. This restriction will continue to apply until the RCS is vented through a passive vent [greater than or equal to] 2.2 in ^{2}. If this passive vent size is established, two charging pumps and one HPSI pump are allowed to be capable of injecting.

The passive vent required if one or two power operated relief valves (PORVs) are inoperable (Technical Specification Action Statements (TSASs) a, b, and c) will be changed from 2.8 in ^{2} or 1.4 in ^{2} to 2.2 in ^{2}. A passive vent of 1.4 in ^{2} is equivalent to the vent area of one PORV. Since the LTOP analysis assumes 2 operable PORVs initially, and then one PORV fails to actuate, RCS overpressure protection will be ensured by a passive vent of 1.4 in ^{2}. However, a passive vent is established by removing a pressurizer PORV or the pressurizer manway, the normal vent path. The value of 2.2 in ^{2} is the minimum size of vent that will ensure RCS pressure remains [less than or equal to] 300 psia, which is more conservative than the Appendix G limits. This vent size will also ensure that RCS pressure does not exceed the SDC System design pressure. In addition, this is the size of vent that will satisfy Technical Specification 3.4.9.1 to allow a 50 °F/hr cooldown rate below 190 °F.

TSAS d will be added to address excessive pumping capacity. The required completion time of ''immediate'' reflects the importance of this restriction, and is consistent with current Technical Specification requirements (Technical Specification 3.1.2.3 TSAS b and Technical Specification 3.5.3 TSAS c) for this situation.

These proposed changes are all more restrictive than the previous requirements, except for allowing 2 charging pumps and one HPSI in Mode 5 between 200 °F and 190 °F, and requiring a vent of 2.2 in ^{2} instead of 2.8 in ^{2} when two PORVs are inoperable. However, the proposed mass input and venting restrictions are consistent with the new LTOP analysis. This analysis has demonstrated that with the proposed restrictions the required LTOP system will provide adequate protection for RCS overpressurization transients. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Increase in Technical Specification Applicability. The applicability of Technical Specification 3.4.9.3 will be expanded to include all of Mode 5, and Mode 6 until the reactor vessel head is removed. The current applicability is limited in Modes 5 and 6 to when the RCS is not vented through a vent [greater than or equal to] 2.8 in ^{2}. Expanding the applicability will ensure an LTOP System is in place, except when RCS pressurization is not possible (reactor vessel head removed). This will ensure the 10 CFR 50 Appendix G limits are not exceeded.

The applicability of Technical Specification 3.4.9.1 will be expanded. The current applicability is Modes 1 through 5. However, concern for non-ductile failure of the reactor vessel and flange applies at all times, not just in Modes 1 through 5. Therefore, the applicability will be expanded. Increasing the applicability of Technical Specification 3.4.9.1 will place additional restrictions on the plant. However, these additional restrictions will ensure the integrity of the RCS, in particular the reactor pressure vessel, is maintained. Therefore, the RCS will continue to function as designed.

These more restrictive changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

LTOP PORV Setpoint Change. The required PORV actuation setpoint will be reduced from [less than or equal to] 450 psig to [less than or equal to] 415 psia (400 psig). The 50 psi setpoint reduction (pressure units have also been changed to agree with control room indication) will cause the PORVs to actuate earlier during an LTOP transient to prevent an RCS overpressurization. It is a more restrictive change that is consistent with the new LTOP analysis.

PORV actuation at the proposed setpoint, in combination with the proposed mass input restrictions, will ensure the 10 CFR 50 Appendix G limits are not exceeded. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

RCP Start Criteria. The requirements to start the first reactor coolant pump (RCP), when RCS temperature is [less than or equal to] 275 °F, will be modified. The new criteria will ensure that starting an RCP will not result in an energy addition transient that could exceed the capability of the steam bubble in the pressurizer to mitigate the event. (No credit for PORV actuation during this energy addition transient was assumed in the new LTOP analysis.) This will ensure that the 10 CFR 50 Appendix G limits are not exceeded.

The proposed RCP restrictions are consistent with the new LTOP analysis. This analysis has demonstrated that with the proposed restrictions the pressurizer will provide adequate protection for RCS overpressurization transients. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Boron Dilution Analysis. The analysis of the boron dilution event contained in the Millstone Unit No. 2 FSAR [Final Safety Analysis Report] Section 14.4.6 assumes that dilution flow rate is limited to 88 gpm in Modes 4, 5, and 6. Since the charging pumps are the assumed dilution source, no more than two charging pumps can be injecting for this assumption to remain valid. This results in a Technical Specification requirement that no more than two charging pumps can be operable when the RCS is in Mode 4 or below (< 300 °F). This requirement will be modified by replacing the word ''operable'' with ''capable of injecting into the RCS.'' This more accurately addresses the boron dilution analysis restriction of limiting the dilution flow to two charging pumps since an inoperable pump can still inject into the RCS. This change is consistent with the boron dilution accident analysis. The boron dilution analysis further assumes that if this dilution flow rate restriction is met, there will be sufficient time for the operators to recognize and terminate the dilution before a complete loss of shutdown margin occurs. Operator action to restore shutdown margin by boration is not assumed.

The proposed changes will not affect the current Technical Specification restriction that no more than two charging pumps can be capable of injecting into the RCS (operable) when the RCS is below 300 °F. However, no corresponding action statement currently exists in Technical Specification 3.1.2.4 to provide guidance if this requirement is not met. The addition of the proposed action statement to Technical Specification 3.1.2.4 will require immediate action to correct this situation. This is consistent with other current Technical Specification requirements (Technical Specification 3.1.2.3 TSAS b and Technical Specification 3.5.3 TSAS c) that address excessive RCS makeup capacity. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

RCS Pressure/Temperature and Heatup/Cooldown Limit Changes. The proposed changes to the heatup and cooldown rates are a result of the new analysis of the RCS pressure/temperature and heatup/cooldown limits. These changes will provide flexibility during plant heatup and cooldown, and especially during equipment manipulations such as securing RCPs, swapping shutdown cooling (SDC) heat exchangers, and initiating SDC.

Figure 3.4.2 will be replaced by two curves, Figures 3.4–2a and 3.4–2b. Each figure will contain the minimum flange boltup temperature and the minimum temperature for criticality. The heatup figure (Figure 3.4–2a) will also contain the inservice leak and hydrostatic testing limits. The temperature change limits will be contained in the new Table 3.4–2, instead of in the LCO [limiting condition for operation]. The new limits will use cold leg temperature instead of average temperature to determine when to change rates. There should be little difference between these two temperatures, and cold leg indication is directly available to the control room operators.

The proposed curves and rates are based on indicated cold leg temperature. This parameter, which is the best available indication of reactor vessel downcomer temperature, will normally be monitored by using either RCS cold leg temperature indication or SDC return temperature. Plant conditions will determine which one is the appropriate indication to use. Actual RCS cold leg temperature will be used if any RCP is operating or natural circulation is occurring. Otherwise, SDC return temperature will be used.

RCP restrictions, assumed in the development of the heatup and cooldown curves, will be added to the curves. Most of the RCP restrictions already exist either in other Technical Specifications or in plant procedures. Two new RCP restrictions will be added to Technical Specifications. The restriction of no more than three RCPs until RCS temperature is above 500 °F already exists in plant procedures, but it will be added to Technical Specifications. The restriction of no more than two RCPs when RCS temperature is below 200 °F already exists in Technical Specifications (3.4.1.4). The RCP restriction of no RCPs below 150 °F during plant cooldown will be added to Technical Specifications. This restriction will have no effect on plant operations because RCPs will normally be secured when cooling down below 150 °F to minimize heat input.

The inservice leak and hydrostatic testing temperature change limit currently specified in Technical Specification 3.4.9.1.c will be relocated to Table 3.4–2. The wording will be modified (clarification only) to specify the limit also applies for one hour prior to the start of inservice leak and hydrostatic testing. This is necessary since the development of the inservice leak and hydrostatic testing test curve assumes isothermal conditions. The wording will also be modified to specify the restrictions apply during testing above the heatup curve instead of above system design pressure. This type of testing is not performed above system design pressure.

The 50 °F/hr cooldown rate and curve will normally be used when the RCS is <190 °F and an RCS vent of >2.2 in^2 has been established. This curve and rate may also be used when RCS cold leg temperature is below 230 °F to demonstrate compliance with Appendix G limits when unanticipated temperature excursions occur.

The current action statements of Technical Specification 3.4.9.1 will be separated by Mode and will be modified. Similar changes will be made to the action statements of Technical Specification 3.4.9.2. A time limit of 72 hours will be placed on the performance of the engineering evaluation. If this evaluation is not performed in this time period, or the evaluation does not allow continued operation, the plant will be required to enter Mode 5 ([less than or equal to] 200 °F), instead of the current requirement to be <200 °F. This slight relaxation will have no significant impact on plant operations because plant temperature is not normally maintained at the mode change temperature limit.

The required RCS pressure will be reduced from 500 psia to 300 psia. This is closer to the actual plant conditions established in Mode 5. The required RCS pressure will be reduced from 500 psig to 500 psia for the Pressurizer, Technical Specification 3.4.9.2. The change in units is consistent with plant instrumentation. Establishing a lower RCS pressure is more conservative because it will result in less pressure stress on either the reactor vessel or the pressurizer.

In other than Modes 1 through 4, immediate action will be required for limit restoration. Violation of these limits is typically more severe when the RCS is cold (<200 °F), therefore an immediate response is appropriate. A time limit of prior to entering Mode 4 will be placed on the performance of the engineering evaluation. This will prevent plant startup until the evaluation has determined that the RCS is acceptable for continued operation.

The frequency of surveillance requirements 4.4.9.1.c, 4.4.9.2, 4.10.3.1 will be increased from once per hour to once per 30 minutes. This more restrictive change will provide the plant operators with earlier indication that a limit may be exceeded, so that action can be taken to prevent exceeding the limit. The proposed changes to the RCS pressure/temperature limits and temperature change rates are based on the new analysis. This analysis uses standard approved methods that ensure the margins of safety required by 10 CFR 50, Appendix G are maintained. The other changes discussed are more restrictive enhancements to Technical Specification requirements. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Other Changes. The scope of the action statement for Technical Specification 3.1.2.6 will be expanded to cover all three flowpaths identified in Technical Specification 3.1.2.2.a. The intent of the current wording is to address all flowpaths. These minor wording changes will meet this intent.

Clarification will be added to SR 4.1.2.8.d to be consistent with SR 4.1.2.7.c. The clarification will allow the boric acid storage tank (BAST) temperature to be verified by checking the ambient air temperature.

A note will be added to Technical Specification 3.4.3 TSAS a to allow the block valve(s) to be cycled during plant cooldown when the block valve(s) is(are) closed due to inoperable PORV(s). The footnote will allow the PORV block valve(s) to be cycled during a plant cooldown to prevent thermal binding. This will ensure the associated block valve(s) can be opened to allow the PORV(s) which is(are) inoperable, can be manually cycled if necessary. Therefore, the PORV block valve(s) will be able to function as designed.

The wording of Technical Specification 3.4.3 Action Statement d will be revised to state what action should be performed, and to remove specific details on how to perform the required action. This does not change the intent of the action statement. Therefore, the pressurizer PORVs will continue to function as designed.

An action statement will be added to Technical Specification 3.4.9.3 to provide an exception to Technical Specification 3.0.4 requirements. This is necessary to allow a plant cooldown to MODE 5 if one or both PORVs are inoperable. MODE 5 conditions may be necessary to repair the PORV(s).

A footnote will be added to Technical Specification 3.5.3 to allow entry into Mode 4 without an operable high pressure safety injection pump. This new footnote will allow the plant to enter Mode 4 where this specification is applicable without any operable HPSI pumps. However, this condition will only be allowed for a very short time period, one hour. The proposed change to Technical Specification 3.4.9.3 will allow a HPSI pump to be operable above 190 °F. However, the 10 °F range before Mode 4 is reached may not allow sufficient time to ensure a HPSI pump is operable. Adding this note will provide the operating crew sufficient time to make an orderly transition into Mode 4. This condition will only be allowed for one hour, which is the same time allowed by the first part of TSAS a for an inoperable HPSI pump.

The LTOP requirements currently contained in Technical Specifications 3.1.2.3 and 3.5.3 will be relocated to the LTOP Technical Specification 3.4.9.3. Relocating requirements within Technical Specifications will not change the technical content of the requirement.

Various redundant or outdated Technical Specification requirements will be eliminated and references will be adjusted to reflect the proposed changes. Removal of redundant or outdated requirements from Technical Specifications and adjustments to references to other requirements will not impact any technical requirements.

Minor wording changes have been made to many of the Technical Specifications contained in this license amendment request. These changes do not change any technical aspect of the Technical Specification affected. They are editorial changes only.

The proposed changes do not alter the way any structure, system, or component functions. There will be no effect on equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will 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 proposed changes will modify the LTOP requirements, RCS pressure/temperature limits, and the RCS heatup and cooldown limits. The majority of the proposed changes are being made as a result of the new pressure/temperature and LTOP analyses performed. The new pressure/temperature curves and heatup and cooldown rates were developed in accordance with the requirements and methods described in 10 CFR 50 Appendix G and are consistent with the criteria contained in the Standard Review Plan Section 5.3.2. The new LTOP mass input and RCP starting restrictions and LTOP PORV setpoints are consistent with the criteria contained in the Standard Review Plan Section 5.2.2. Additional changes have been proposed to correct various items identified during the review of the Millstone Unit No. 2 Technical Specifications. The proposed changes do not change the requirements to maintain RCS pressure and temperature within the requirements defined in Technical Specifications. This will ensure the integrity of the reactor vessel is maintained during all aspects of plant operation. Therefore, there is no significant effect on the probability or consequences of any accident previously evaluated and no significant impact on offsite doses associated with previously evaluated accidents. This License Amendment Request does not result in a reduction of the margin of safety as defined in the Bases for the Technical Specifications addressed by the proposed changes.

The NRC has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC [significant hazards consideration]. The changes proposed herein to correct terminology, numbering, references, and relocating requirements within Technical Specifications are enveloped by example (i), a purely administrative change to Technical Specifications. The more restrictive changes proposed herein that are based on the new analyses performed and the more restrictive enhancements are enveloped by example (ii), a change that constitutes an additional limitation, restriction, or control not presently included in Technical Specifications. All other changes proposed herein are not enveloped by a specific example.

As described above, this License Amendment Request does not impact the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO [Northeast Nuclear Energy Company] has concluded that the proposed changes do not involve an SHC.

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.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Deputy Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut.

Date of amendment request: December 8, 1997.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to resolve several compliance issues. The proposed changes would (1) correct the wording and the formula in TS Definition 1.18 ''Azimuthal Power Tilt—T_q;'' (2) correct the wording in TS 4.1.1.1.2 ''Reactivity Control Systems Shutdown Margin—T_{avg} [less than or equal to] 200 °F;'' (3) correct the mode applicability from Mode 3 to Modes 1 and 2 in TS 3.1.3.4 ''Reactivity Control Systems—Rod Drop Time;'' (4) correct the terminology used to refer to the power dependent insertion limit alarm in TS 4.1.3.6 ''Reactivity Control Systems—Regulating CEA [Control Element Assembly] Insertion Limits;'' (5) add a footnote for Mode 4 operability requirement clarification to TS 3.5.3 ''Emergency Core Cooling Systems, ECCS Subsystems—T_{avg} <300 °F;'' (6) correct the wording, frequency, and reference number for the surveillance requirements in TS 3.6.3.2 ''Containment Systems Containment Ventilation System;'' (7) correct the nomenclature used for the A.C. busses in TSs 3.8.2.1 and 3.8.2.1A ''Onsite Power Distribution Systems A.C. Distribution—Operating;'' (8) correct TS Bases by modifing the applicable sections to reflect the proposed changes; (9) delete the word ''original'' from the statement ''original design provision'' in Design Features Section—TSs 5.1.3 ''Flood Control,'' 5.2.3 ''Penetrations,'' 5.3.2 ''Control Element Assemblies,'' and 5.7.1 ''Seismic Classification;'' and (10) delete Design Features Section—TS 5.9 ''Shoreline Protection.''

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

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

The proposed change in the wording and associated formula of Technical Specifications Definition 1.18 will ensure the calculated value of Azimuthal Power Tilt (T_q) used to verify compliance with Technical Specification 3.2.4 is associated with the quadrant of highest power production with respect to the average of the four quadrants, instead of the quadrant that deviates the most (increases or decreases) from the average of the four quadrants. This is consistent with the method by which power distribution factors are calculated and applied in the accident analysis and how the Core Power Distribution Monitoring System calculates T_q. The proposed change will not alter the way T_q is calculated by the Core Power Distribution Monitoring System, nor will it alter any of the power distribution assumptions used in the accident analysis. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Surveillance Requirement (SR) 4.1.1.1.2 requires that the difference between predicted and measured core reactivity values be maintained within [plus or minus] 1.0% [delta]k/k, and that an adjustment be made between the measured and predicted core reactivity conditions prior to exceeding 60 EFPD [effective full power days] following a refueling outage. The proposed change will not affect the requirement to maintain predicted and measured core reactivity values within [plus or minus] 1.0% [delta]k/k. However, it will no longer be necessary to make an adjustment prior to exceeding 60 EFPD provided the [plus or minus] 1.0% [delta]k/k requirement is met. Historically, this difference has been small at Millstone Unit No. 2 (less than approximately [plus or minus] 1.0% [delta]k/k) and an adjustment has not been necessary to ensure the [plus or minus] 1.0% [delta]k/k requirement is met. The fact that no adjustment (normalization) will be necessary when reactivity differences are small will not affect the ability to identify reactivity anomalies. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to Technical Specification 3.1.3.4 will change the applicability from Mode 3 to Modes 1 and 2. This is necessary to allow performance of SR 4.1.3.4 at the conditions in the accident analysis, and also specified in the [Limiting] Condition [for] Operation (LCO). CEA [Control Element Assembly] drop time is important for the mitigation of accidents that are initiated while the reactor is critical. To ensure the CEA drop time assumed in the accident analysis is valid, it is necessary to verify CEA drop time with plant conditions consistent with those expected when the reactor is critical. This proposed change will allow this verification, and thereby ensure the CEAs will function as designed to mitigate design basis accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to SR 4.1.3.6 will modify the terminology used to refer to the power dependent insertion limit (PDIL) alarm to agree with plant terminology. This change will not alter equipment operation or any technical aspect of the SR. The information added to the Bases will specify what equipment provides the PDIL alarm. These changes will eliminate any confusion with alarm terminology. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3.5.3 requires an operable flowpath capable of taking a suction from the refueling water storage tank (RWST) on a safety injection actuation signal (SIAS), and automatically transferring suction to the containment sump on a sump recirculation actuation signal (SRAS) in Mode 4. In Mode 4, the automatic SIAS generated by low pressurizer pressure and high containment pressure, and the automatic SRAS generated by low RWST level, are not required to be operable. Automatic actuation in Mode 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Since the manual actuation (trip pushbuttons) portions of the safety injection and sump recirculation actuation signal generation are required to be operable in Mode 4, credit can be taken for remote manual operation to generate the SIAS and SRAS which will position all components to the required accident position. The proposed change to Technical Specification 3.5.3 will add a footnote (***) to explain how these requirements are met in Mode 4. This change will not reduce operability or surveillance requirements for the Emergency Core Cooling System (ECCS) subsystem required to be operable by Technical Specification 3.5.3. The ECCS will continue to function as designed to mitigate design basis accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to Technical Specification 3.6.3.2 will revise the wording of the LCO and SR by changing ''locked closed'' to ''sealed closed,'' and deleting the requirement to be electrically deactivated. The action statement will also be revised to reflect these proposed changes. These changes will not affect the requirement for the containment purge valves to be closed in Modes 1 through 4. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to SR 4.6.1.7 will change the surveillance frequency from ''prior to each reactor startup'' to ''at least once per 31 days.'' This change, which will require the surveillance to be performed more often (assuming a normal plant startup sequence) will provide additional assurance that the containment purge valves are sealed closed. In addition, this change will ensure consistency between the SR and the applicability of this specification, and also with the requirements to verify containment integrity in accordance with Technical Specification 3.6.1.1. Therefore, the proposed change will not significantly increase the probability or consequences of an accident previously evaluated.

The change in numbering of SR 4.6.1.7 to SR 4.6.3.2 is an administrative change only. It will not affect any technical aspect of the SR. Therefore, the proposed change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes to Technical Specifications 3.8.2.1 and 3.8.2.1A will modify the nomenclature used to refer to the vital A.C. buses to be consistent with the terminology used by Operations Department personnel and the nomenclature contained in their procedures. These changes will not alter equipment operation or any technical aspects of these specifications. These proposed changes are administrative changes only. The A.C. buses will continue to function as designed to mitigate design basis accidents. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes to Technical Specifications 5.1.3, 5.2.3, 5.3.2, and 5.7.1 will remove the word ''original.'' Reference to original design is not appropriate since these items can be changed by approved processes. However, these changes will still require the items addressed by these specifications to be designed and maintained in accordance with the Final Safety Analysis Report (FSAR). The proposed changes have no affect on the current approved plant design. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 5.9 will be deleted. The required provisions for shoreline protection have been completed, and this Technical Specification is no longer necessary. The removal of this outdated specification will not impact any current requirements. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will 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 proposed change to the definition of T_q will make the Technical Specification definition consistent with the approved calculation methodology. This will ensure the core power distribution is consistent with accident analysis assumptions. The proposed change to the wording of SR 4.1.1.1.2 will not affect the acceptance criteria of [plus or minus] 1.0% [delta]k/k, which ensures the accident analysis accurately reflects core reactivity conditions. The proposed change in the applicability of Technical Specification 3.1.3.4 will allow verification of CEA drop time at plant conditions assumed in the accident analysis. This will ensure the CEAs will function as assumed. The proposed change to SR 4.1.3.6 will modify the terminology used to refer to the PDIL alarm to agree with plant terminology. This change will not alter equipment operation or any technical aspect of the SR. Adding the footnote to Technical Specification 3.5.3 will not change any technical aspects of this specification. One ECCS subsystem will be available for accident mitigation. The proposed change in wording of Technical Specification 3.6.3.2 will not affect the requirement for the containment purge valves to be closed in Modes 1 through 4. The proposed change in the frequency of performance for SR 4.6.1.7 will provide greater assurance that the containment purge valves are closed to prevent the potential release of radioactive material through these penetrations during accident conditions. The proposed changes in terminology in Technical Specifications 3.8.2.1 and 3.8.2.1A will not change any technical requirements for the equipment covered. The equipment will still function as assumed. Modifying the Bases of Technical Specifications are necessary to be consistent with the proposed changes will not change any requirements of these specifications. The modification to Technical Specifications 5.1.3, 5.2.3, 5.3.2, and 5.7.1 will not affect the requirement to maintain these items in accordance with requirements contained in the FSAR. Deleting Technical Specification 5.9 will not affect any requirements since the requirements contained in this specification have already been completed.

The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, these proposed changes will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

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.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Deputy Director: Phillip F. McKee.

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: December 12, 1997.

Description of amendment request: The proposed amendment would revise the facility Technical Specifications (TSs) regarding normal working hours of plant staff to provide for shift duration of 12 hours. It would also revise the TSs to maintain existing ''once per shift'' surveillance requirements at 8-hour intervals.

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

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

Response:

Establishing operating personnel work hours at ''a normal 8 to 12 hour day, nominal 40-hour week'' allows normal plant operations to be managed more effectively and does not adversely affect performance of operating personnel. Overtime remains controlled by site administrative procedures in accordance with NRC Policy Statement on working hours (Generic Letter 82–12). If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. No physical plant modifications are involved and none of the precursors of previously evaluated accidents are affected. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Editorial changes clarify sections 6.2.2.6.b. and 6.2.2.6.c. without changing the intent or meaning. [...] Changes to sections 4.5.F.3., 4.5.F.4., 4.5 Bases, and 4.7.A.7.a. do not change the intent or meaning of the Technical Specifications, do not change operating procedures, and are consistent with surveillance requirements. [Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.]

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

Establishing operating personnel work hours at ''a normal 8 to 12 hour day, nominal 40-hour week'' allows normal plant operation to be managed more effectively and does not adversely affect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82–12). No physical modification of the plant is involved. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, plant operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

Editorial changes clarify sections 6.2.2.6.b. and 6.2.2.6.c. without changing the intent or meaning. [* * *] Changes to sections 4.5.F.3., 4.5.F.4., 4.5 BASES, and 4.7.A.7.a. do not change the intent or meaning of the Technical Specifications or operating procedures. All previously performed functions are being maintained.

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

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

Establishing operating personnel work hours at ''a normal 8 to 12 hour day, nominal 40-hour week'' allows normal plant operations to be managed more effectively and does not adversely affect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82–12). The proposed change involves no physical modification of the plant, or alterations to any accident or transient analysis. [* * *] Therefore, the change does not involve any significant reduction in a margin of safety.

Editorial changes clarify sections 6.2.2.6.b. and 6.2.2.6.c. without changing the intent or meaning. [* * *] Changes to sections 4.5.F.3., 4.5.F.4., 4.5 BASES, 4.7.A.7.a. do not change the intent or meaning of the Technical Specifications or operating procedures.

All previously performed functions are being maintained. Therefore, the changes do not involve any 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: December 19, 1997.

Description of amendment request: The proposed amendment would revise the Hope Creek Generating Station (HCGS) Technical Specifications (TS) to incorporate changes that reflect the completion of the Salt Drift Monitoring Program.

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

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

The changes, which update the Terrestrial Ecology Monitoring Program status, are administrative in nature and in no way affect the initial conditions, assumptions, or conclusions of the Hope Creek Generating Station accident analyses. In addition, the proposed changes would not affect the operation or performance of any equipment assumed in the accident analyses. Based on the above information, we conclude that the proposed changes would not significantly increase the probability or consequences of an accident previously evaluated.

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

As previously stated, the proposed changes are administrative in nature and in no way impact or alter the configuration or operation of the facilities and create no new modes of operation. PSE&G therefore concludes that the proposed changes would not create the possibility of a new or different kind of accident.

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

The changes are administrative in nature and in no way affect plant or equipment operation or the accident analysis. PSE&G therefore concludes that the proposed changes would not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: November 14, 1997.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) to provide surveillance requirements for the service water accumulator vessels. Specifically, surveillance requirements are provided for vessel level, pressure and temperature, and discharge valve response time. The surveillance requirements are included in TS 3/4.6.1.1 and 3/4.6.2.3, and the applicable Bases sections are expanded to provide supporting information.

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 changes provide surveillance requirements for the Service Water [SW] accumulator tank level, pressure and temperature parameters and the discharge valve response time test. Supporting information is included in the Bases section of the applicable technical specifications. The SW accumulator tank and discharge valve design has been reviewed and approved by the NRC staff as documented in NRC Safety Evaluation Report (SER) dated June 19, 1997. The proposed surveillance requirements do not alter the design as reviewed by the NRC staff. The addition of tank parameter surveillance requirements to the technical specifications does not alter the physical plant arrangement or the installed monitoring instrumentation. The proposed addition of tank discharge valve response time surveillance requirements to the technical specifications does not alter the method of performing these surveillance requirements.

Therefore the proposed changes do not increase the probability of an accident. The surveillance requirements provide additional controls for ensuring the SW accumulator tank and discharge valves will be maintained within the design parameters assumed in the safety analysis. This provides added assurance that the accumulator tanks and discharge valves will be capable of performing their required design function during accident conditions. There is no change to the performance requirements of these components in preventing two phase flow conditions and water column separation waterhammer vulnerabilities identified in GL [Generic Letter] 96–06. Therefore, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

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

The proposed changes provide surveillance requirements for Service Water Accumulator tank level, pressure and temperature and discharge valve time response. Supporting information is included in the Bases section of the applicable technical specifications. The SW accumulator tank and discharge valve design has been reviewed and approved by the NRC staff as documented in NRC Safety Evaluation Report (SER) dated June 19, 1997. The proposed surveillance requirements do not alter the plant configuration. Installed instrumentation will be used to accomplish the tank surveillance requirements. The current plant installation also provides for completion of the discharge valve response time surveillance utilizing test equipment in accordance with plant procedures and configurations. Therefore the performance of these surveillance requirements does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The Service Water Accumulator Vessels and discharge valves were installed to address the Generic Letter 96–06 issues of column separation waterhammer and two phase flow in the containment fan coil unit (CFCU) piping during an accident involving loss of offsite power. This design has been reviewed and approved by the NRC staff as documented in NRC Safety Evaluation Report (SER) dated June 19, 1997. The proposed surveillance requirements do not alter the design as reviewed by the NRC staff. By providing added assurance that these components are capable of performing their specified safety function as assumed in the safety analysis, the additional surveillance requirements assure system operability to further minimize the possibility of waterhammer and two phase flow in the CFCU piping during accident conditions. The proposal therefore minimizes the possibility of a new or different kind of accident from those previously evaluated accidents.

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

The additional surveillances provide added assurance that the margin of safety assumed in the containment integrity and containment cooling technical specification will be maintained. The additional surveillance requirements further ensure that in the event the SW accumulator vessels are out of specification or the discharge valves do not meet their response time requirements, corrective actions will be completed in accordance with the existing containment integrity technical specification allowed outage time to restore containment integrity. The surveillance requirements further ensure that in the event the SW accumulator vessel or discharge valves do not meet these requirements, corrective actions will be completed in accordance with the containment cooling technical specification allowed outage time to restore the full complement of containment fan coil units to operability. Since the proposal maintains the margin of safety provided in the containment integrity and containment cooling technical specification, there is no 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.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

NRC Project Director: John F. Stolz.

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

Date of amendment requests: September 16, 1997.

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.4.13, ''RCS Operational Leakage,'' TS 5.5.2.11, ''Steam Generator (SG) Tube Surveillance Program,'' and TS 5.7.2, ''Special Reports.'' The proposed change is to allow steam generator tube repair using ASEA Brown Boveri/Combustion Engineering (ABB/CE) leak tight sleeving as an alternative steam generator tube repair to plugging.

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 supporting technical evaluation and safety evaluation for the ASEA Brown Boveri/Combustion Engineering (ABB/CE) leak tight sleeves demonstrate that the sleeve configuration will provide steam generator (SG) tube structural and leakage integrity under normal operating and accident conditions. The sleeve configurations have been designed and analyzed in accordance with the requirements of the ASME Code. Mechanical testing has shown that the sleeve and sleeve joints provide margin above acceptance limits. Ultrasonic Testing (UT) is used to verify the leak tightness of the weld above the tubesheet. Testing has demonstrated the leak tightness of the hardroll joint due to the reinforcing effect of the tubesheet. Tests have demonstrated that tube collapse will not occur due to postulated Loss of Coolant Accident (LOCA) loadings.

A new, more conservative, Technical Specification (TS) SG tube leakage rate requirement is introduced by this change. Accident analysis assumptions remain unchanged in the event that significant leakage does occur from the sleeve joint or that the sleeve assembly ruptures. Any leakage through the sleeve assembly is fully bounded by the existing SG tube rupture analysis included in the San Onofre Nuclear Generating Station (SONGS) Updated Final Safety Analysis Report. Reactor coolant flow reduction from sleeving is addressed by a ratio of number of tubes sleeved to equal a plugged tube. The proposed sleeving repair process does not adversely impact any other previously evaluated design basis accidents.

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

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

Installation of the sleeves does not introduce any significant changes to the plant design basis. The use of a sleeve to span the area of degradation of the SG tube restores the structural and leakage integrity of the tubing to meet the original design bases. Stress and fatigue analysis of the sleeve assembly shows that the requirements of the ASME Code are met. Mechanical testing has demonstrated that margin exists above the design criteria. Any hypothetical accident as a result of any degradation in the sleeved tube would be bounded by the existing tube rupture accident analysis.

Therefore, the operation of the facility in accordance with proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The use of sleeves to repair degraded SG tubing has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of the ASME Code and draft Regulatory Guide (RG) 1.121 and to maintain the primary to secondary pressure boundary under normal and postulated accident conditions. The safety factors used in the verification of the strength of the sleeve assembly are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in SG design. The operational and faulted condition stresses and cumulative usage factors are bounded by the ASME Code requirements. The sleeve assembly has been verified by testing to prevent both tube pullout and significant leakage during normal and postulated accident conditions. A test program was conducted to ensure the lower hardrolled joint design was leak tight and capable of withstanding the design loads. The primary coolant pressure boundary of the sleeve assembly will be periodically inspected by Non-Destructive Examination to identify sleeve degradation due to operation.

Installation of the sleeves will decrease the number of tubes which must be taken out of service due to plugging. There is a small amount of primary coolant flow reduction due to the sleeve for which the equivalent sleeve to plug ratio is assigned based on sleeve length. The ratio is used to assess the final equivalent plugging percentage as an input to other safety analyses. The sleeve maintains the design basis requirements for the SG tubing.

Therefore, operation of the facility with the proposed changes 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison

Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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

Date of amendment requests: October 17, 1997.

Description of amendment requests: The licensee proposes to amend the licenses for SONGS Units 2 and 3 to revise the Final Safety Analysis Report (FSAR) to permit digital radiation monitor installation for both trains supplying the containment purge isolation signal, and permit digital radiation monitor installation for both trains supplying the control room isolation signal.

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

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

The proposed change is required to permit using digital radiation monitors as input to both trains of the Control Room Isolation Signal (CRIS), and to both trains of the Containment Purge Isolation Signal (CPIS). These changes will allow replacement of the remaining safety related obsolete radiation monitor equipment to address spare parts and equipment availability issues. The new containment airborne radiation digital monitor will have the same basic architecture as the existing analog system, and serves to perform the same function. In addition, the digital radiation monitors are expected to be more reliable than the existing equipment which is of an analog design.

Furthermore, defense-in-depth equipment is available that either provides, or allows for, actions to mitigate the release of offsite and Control Room doses to within existing licensing limits based on realistic event input assumptions. Analyses show that if ''realistic'' input assumptions are utilized and reasonable operator actions are allowed, then acceptable dose consequences result both to the general public offsite, and to the Control Room operators.

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

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

The proposed change will permit upgrading the existing analog radiation monitors with upgraded digital radiation monitors. Replacement of an analog system to a predominantly digital system, uses software algorithms to perform the required functions. A satisfactory software verification and validation (V&V) report, including continued software change control procedures, provides assurance that a software common mode failure is not likely.

In addition, the design, installation, testing, maintenance, and operation of the affected equipment will assure that no new or different kinds of accidents will be created. The ESFAS radiation monitors involved are portions of systems that respond to accidents. They can not, by their actions or inactions, create a new or different accident from any accident 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 proposed change does not involve a significant reduction in a margin of safety.

The CRIS and CPIS Radiation Monitor Systems provide an accident mitigation function for offsite doses (10 CFR 100) and Control Room doses (10 CFR 50 Appendix A, General Design Criteria 19). A change in the margin of safety is introduced due to the possibility of a software common mode failure in redundant equipment simultaneously affecting equipment performing a different function.

This change is not a significant reduction in the margin of safety, however, due to the following:

(1) A probabilistic risk analysis has determined that the availability of the affected radiation monitors, including software, should be better than the existing equipment based on industry data to date,

(2) The software V&V and preoperational testing to be performed will provide assurance of system operation, and

(3) The combined occurrence of a software common mode failure that simultaneously causes failure of all available ESFAS radiation monitors concurrent with a design bases accident is very unlikely.

In the unlikely event of a software common mode failure that causes all ESFAS radiation monitors to be inoperable concurrent with a design bases accident, analyses show that if ''realistic'' input assumptions are utilized and reasonable operator actions are allowed, then acceptable dose consequences result both to the general public offsite, and to the Control Room operators.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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 amendment request: December 18, 1997.

Description of amendment request: The proposed amendments would modify or delete obsolete conditions from the Unit 1 and Unit 2 Operating Licenses. The changes are editorial or administrative in nature.

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

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

The proposed changes either remove or modify provisions in the Plant Hatch Unit 1 and Unit 2 Operating Licenses that have been completed or are otherwise obsolete. Certain Surveillance Requirements (SRs) that were either added or modified at the time of Improved Technical Specifications (ITS) implementation were listed in the Operating Licenses with a schedule for performance. With the exception of Unit 1 SR 3.8.1.18, all SRs are deleted from the Operating Licenses, because they have since been performed according to schedule, and will henceforth be performed in accordance with the Technical Specifications.

A requirement for submittal of the Unit 1 inservice inspection plan for the recirculation and residual heat removal systems' piping is deleted due to completion of the activity.

Two exemptions granted at Unit 2 startup are deleted due to completion of the required activities associated with the exemptions. These were seismic qualification demonstration for the Unit 2 reactor protection system power supply and completion of the long-term BWR [boiling water reactor] Owner's Group Mark I containment program.

A requirement to conduct the Unit 2 Initial Test Program according to the requirements in Chapter 14 of the Final Safety Analysis Report without major changes is deleted due to completion of the activity. A condition relating to environmental protection is deleted from the Unit 2 Operating License, since it was superseded by the Environmental Protection Plan (Nonradiological), Appendix B to the Operating Licenses. Attachment 2, Items To Be Completed Prior To Opening Main Steam Isolation Valves, is deleted due to completion of the activities.

The proposed changes discussed above are strictly administrative/editorial and do not affect the operation or function of any plant system, component, or structure. Therefore, the proposed changes do not increase the probability of occurrence or the consequences of a previously evaluated accident.

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

The proposed administrative/editorial changes do not alter the operation of any plant system or equipment and do not introduce a new mode of operation. Thus, the proposed changes cannot create a new accident initiating mechanism. Therefore, the proposed changes do not create the possibility of a new and different type of accident from any previously evaluated.

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

Since the proposed changes are strictly administrative/editorial and do not involve any physical or procedural changes to the plant, the margin of safety, as defined in the bases for any Technical Specification is not affected by the proposed changes.

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.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC.

NRC Project Director: Herbert N. Berkow.

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: December 17, 1997.

Description of amendment request: The proposed amendment would extend the surveillance interval of the containment spray system nozzle air flow test from five years to ten years.

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

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

The proposed change does not result in any hardware changes. The Containment Spray system trains or nozzles are not assumed to be the initiators of any analyzed events. Extending the surveillance interval for performing the Containment Spray system nozzle air flow test from five to ten years does not represent a significant increase in the probability of an accident. The Containment Spray system nozzles are not precursors to any accident analyses.

The Containment Spray system trains and nozzles function to mitigate the consequences of an analyzed event by providing spray flow to containment during an accident. The proposed change still provides assurance that the Containment Spray system nozzles will be maintained operable due to the passive nature of the design, the materials of construction, and the low-stress non-wetted environment. The extension of the surveillance interval does not significantly increase the probability or consequences of an accident since the nozzle will still be OPERABLE between surveillance tests.

B. Operation of the facility in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not necessitate a physical alteration of the plant or changes in parameters governing normal plant operation. No new or different types of equipment will be installed. The proposed change will still ensure Containment Spray system nozzle OPERABILITY is adequately maintained.

C. Operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

The increased interval between the Containment Spray system nozzle air flow test is acceptable due to the passive design of the nozzles and industry operating experience as detailed in NURG–1366. The increased interval is considered acceptable for maintaining nozzle OPERABILITY. The Containment Spray system, including the nozzles, will continue to provide their required safety function with the increase from five to ten years between inspections.

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

Local Public Document Room location: Wharton County Junior College, J.M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036–5869.

NRC Project Director: John Hannon.

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: December 31, 1997.

Description of amendment request: The proposed amendment would revise Technical Specifications 2.1 (Safety Limits), 2.2 (Limiting Safety System Settings), and 3/4.2.5 (Departure from Nucleate Boiling Parameters) by including alternate operating criteria to allow continued plant operation with a reduced measured reactor coolant system flow rate, if necessary.

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

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

The affected Reactor Protection System functions will continue to provide their current safety function under alternate operating criteria for reduced measured Reactor Coolant System flow conditions. The OT Delta-T [Overtemperature Delta-T], OP Delta-T [Overpower Delta-T], and f(Delta-I) [a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers] safety-analysis reactor trip setpoints have been recalculated to appropriately reflect the reduced flow conditions. In doing so, the difference, or margins, between the nominal and maximum values of the reference trip setpoints (*i.e.*, K1, and K4 for the OT Delta-T and the OP Delta-T setpoints, respectively) have been maintained so that the Total Allowance remains unchanged and, therefore, the instrument accuracy uncertainties are unaffected.

Furthermore, implementation of the provisions for reduced measure Reactor Coolant System flow under alternate operating criteria for the South Texas Project Technical Specifications does not increase the probability or consequences of an accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. This change cannot directly initiate an accident. The consequences of accidents previously evaluated in the UFSAR are unaffected by this proposed change because no change to any equipment response or accident mitigation scenario has resulted. There are no additional challenges to fission product barrier integrity. Therefore, the probability of an accident previously evaluated has not been increased.

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

No new failure mechanisms or accident scenarios or limiting single failures are introduced as a result of this proposed change. Operation of the plant will be consistent with that previously modeled. All of the accident analyses previously evaluated in the UFSAR for South Texas Project Units 1 and 2 have been evaluated to support alternate operating condition with a 3 percent reduction in the minimum measured Reactor Coolant System flow. The new nominal Reactor Coolant System operating conditions supported by these evaluations have been determined. Revised Core Thermal Safety Limits have been established and will be incorporated into the Technical Specifications for the 3 percent Reactor Coolant System measured flow reduction; and, the OT Delta-T and OP Delta-T setpoints are re-calculated based on the new Safety Analysis Limits, appropriate for the reduced flow operation. These reactor protection system functions affected by the change in operating conditions will, therefore, continue to provide an appropriate response equivalent to current safety analysis modeling. The proposed Technical Specification amendment does not challenge the performance or integrity of safety-related systems. The possibility of a new or different kind of accident, therefore, is not created.

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

The modification will have no effect on the availability, operability, or performance of the South Texas Project safety-related systems and components. This is based on: the evaluation performed of all accidents previously evaluated in the UFSAR for operation of South Texas Project Units 1 and 2 at reduced Reactor Coolant System flow conditions; establishment of revised Core Thermal Safety Limits that are reflected in the proposed Technical Specification applicable for the 3 percent Reactor Coolant System flow reduction; and, the appropriately re-calculated OT Delta-T and OP Delta-T setpoints, also applicable for these reduced flow conditions. Allowing provision for these alternate operating criteria does not prevent inspections or surveillance required by the Technical Specifications. The margin of safety associated with the acceptance criteria for any accident is unchanged, and therefore, the proposed modification will not reduce the margin of safety as defined in the Bases of the South Texas Project Technical Specifications.

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

Local Public Document Room location: Wharton County Junior College, J.M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036–5869.

NRC Project Director: John N. Hannon.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: December 23, 1997.

Description of amendment request: The license amendment request proposes changes to technical specification surveillances to remove the requirements related to accelerated testing of the standby emergency diesel generators, consistent with the recommendations in NRC Generic Letter 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators.''

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 changes do not significantly increase the probability of occurrence of a previously evaluated accident because the standby diesel generators (including the High Pressure Core Spray [HPCS] diesel generator) are not initiators of previously evaluated accidents. The standby diesel generators mitigate the consequences of previously evaluated accidents involving a loss of offsite power. The Perry Nuclear Power Plant (PNPP) program developed to meet the Maintenance Rule (10 CFR 50.65) will continue to ensure the diesel generators perform their function when called upon. The change to the surveillance frequency does not affect the design of the diesel generators, the operational characteristics of the diesel generators, the interfaces between the diesel generators and other plant systems, the function, or the reliability of the diesel generators. Thus, the diesel generators will be capable of performing their accident mitigation function, there is no impact to the radiological consequences of any accident analysis, and the probability and consequences of previously evaluated accidents are not increased by this activity.

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

The proposed activity involves a change to the frequency for specific technical specification surveillance requirements. No physical or operational changes to the diesel generators or supporting systems are made by this activity. Since the proposed changes do not involve a change to the plant design or operation and thus no new system interactions are created by this change, these changes do not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated in the Updated Safety Analysis Report. The proposed changes only address the methods used to ensure diesel generator reliability. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes involve the methods used to ensure diesel generator performance and reliability. No changes, other than to frequency, are made to Technical Specification Surveillance Requirements 3.8.1.2 and 3.8.1.3. The NRC, in Generic Letter 94–01, has acknowledged the acceptability of the use of the Maintenance Rule program for the diesel generators to ensure diesel generator performance in lieu of accelerated testing. These proposed changes do not involve a change to the plant design or operation, and thus do not affect the design of the diesel generators, the operational characteristics of the diesel generator, the interfaces between the diesel generators and other plant systems, or the function or reliability of the diesel generators. Because the diesel generator performance and reliability will continue to be ensured by the diesel generator program to meet the Maintenance Rule, the proposed changes do not result in a significant reduction in the margin of safety.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Project Director: Richard P. Savio.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: December 23, 1997.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 4.4.5, ''Reactor Coolant System—Steam Generators—Surveillance Requirements (SRs).'' SR 4.4.5.8 would be modified to provide flexibility in the scheduling of steam generator inspections during refueling outages.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed. The proposed change to Technical Specification (TS) Surveillance Requirement (SR) 4.4.5.8, to allow performance of required visual inspections of the secured internal auxiliary feedwater header, header to shroud attachment welds, and the external header thermal sleeves during the third period of the ten-year Inservice Inspection Interval, does not affect any Updated Safety Analysis Report (USAR) accident initiators. These inspections will continue to take place at a prescribed time interval scheduled similar to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI components. Therefore, it can be concluded that the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed change does not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not alter the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed changes.

These inspections were established to ensure that there are no new failure mechanisms resulting from these components. These inspections will continue to take place in the third period of each inservice inspection interval. No new or different types of failures or accident initiators are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because visual inspections will be performed on a prescribed frequency that is consistent with the schedules established for ASME Code components in accordance with ASME Code Section XI.

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.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: December 23, 1997.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 1.0, ''Definitions,'' to clarify the meaning of core alteration; would relocate TS Section 3/4.9.5, ''Refueling Operations—Communications,'' and the associated bases to the Technical Requirements Manual; and would add TS Section 3.0.6 and the associated bases to address the return to service of inoperable equipment.

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

The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the probability of previously analyzed accidents is not affected by the criteria in the core alteration definition (Technical Specification (TS) 1.12). Nor do these changes, the proposed relocation of the refueling communications TS 3/4.9.5 and Bases to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM), or the proposed addition of new TS 3.0.6 and Bases regarding return to service of inoperable equipment, affect any accident initiator, or assumption made in any safety analysis. The proposed changes are administrative in nature and are consistent with NUREG–1430, Revision 1, ''Standard Technical Specifications, Babcock and Wilcox Plants,'' dated April 1995, as modified by a pending NUREG–1430 change approved by the NRC, Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler Number 165.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not significantly alter the source term, containment isolation, or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated. No new or different types of failures or accident initiators are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because no inputs into the calculation of any Technical Specification Safety Limit, Limiting Safety System Settings, Technical Specification Limiting Condition for Operation, or other previously defined margins for any structure, system, or component important to safety are being affected by the proposed changes.

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.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH.

Attorney for licensee: Jack Newman, Al Gutterman, Morgan, Lewis & Brockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Acting Project Director: Richard P. Savio.

Yankee Atomic Electric Company, Docket No. 50–029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: December 18, 1997.

Description of amendment request: By letter dated May 15, 1997, the licensee submitted a License Termination Plan. The NRC previously published a notice dated August 14, 1997, in the **Federal Register** (62 FR 43559) advising of receipt of the Plan. The proposed request is for a license amendment approving the Plan for the Yankee Nuclear Power Station.

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:

The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. Accident analyses are included in the approved Decommissioning Plan and incorporated into the FSAR. All decommissioning and fuel storage activities described in the License Termination Plan are consistent with those in the approved Decommissioning Plan. No systems, structures, or components that could initiate or be required to mitigate the consequences of an accident are affected by the proposed change in any way not previously evaluated in the approved Decommissioning Plan. Therefore, the proposed change is administrative in nature and does not involve an increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated. Accident analyses are included in the approved Decommissioning Plan and are incorporated into the FSAR. All decommissioning and fuel storage activities described in the License Termination Plan are consistent with those in the approved Decommissioning Plan. The proposed change does not affect plant systems, structures, or components in any way not previously evaluated in the approved Decommissioning Plan, and no new or different failure modes will be created. Therefore, the proposed change is administrative in nature and does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety. Approval of the License Termination Plan by license amendment is administrative in nature since all decommissioning and fuel storage activities described in the License Termination Plan are consistent with those in the approved Decommissioning Plan. 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.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: Seymour H. Weiss.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action, see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

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

Date of application for amendments: October 2, 1997.

Brief description of amendments: The amendment changes the Calvert Cliffs Unit 1 Technical Specification Requirements 4.8.1.1.2.a.5, 4.8.1.1.2.d.4, and 4.8.1.1.2.d.5. Baltimore Gas and Electric Company is planning to modify existing 1B emergency diesel generator (EDG) to increase its rated continuous capacity from 2700 kW to 3000 kW by increasing the mechanical capacity of the engine. The change revises the above surveillance requirements to reflect the new electrical capacity of 1B EDG.

Date of issuance: January 5, 1998.

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

Amendment No.: 224.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59913).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated January 5, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

Date of amendment request: September 25, 1997.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by modifying the Limiting Condition for Operation (LCO) 3.6.1.2 (Containment Leakage), the associated action, and Surveillance Requirement (SR) 4.6.1.2 for Waterford Steam Electric Station, Unit 3 (Waterford 3). The air lock door seal leakage rate acceptance criteria in TS 6.15 is being changed from $0.01L_a$ to $0.005L_a$. TS 6.15 is also being modified to make the terms used in the Containment Leakage Rate Testing Program consistent with terms used in the TS.

Date of issuance: January 15, 1998.

Effective date: January 15, 1998.

Amendment No.: 138.

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54872).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 15, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

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

Date of application for amendment: October 10, 1997.

Brief description of amendment: The amendment revises the Oyster Creek Nuclear Generating Station (OCNGS) operating license and technical specifications to reflect the registered trade name of ''GPU Energy'' under which the owner of OCNGS now does business and to reflect the change of the legal name of the operator of OCNGS from GPU Nuclear Corporation to GPU Nuclear, Inc. In addition, two minor editorial corrections associated with the name change are included in the amendment.

Date of issuance: January 14, 1998.

Effective date: As of the date of issuance, with full implementation within 30 days.

Amendment No.: 194.

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

Date of initial notice in **Federal Register**: November 5, 1997 (62 FR 59915). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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 application for amendments: October 8, 1997, and October 21, 1997.

Brief description of amendments: The amendments increase both the minimum required ice mass per ice basket and the total minimum required ice mass in the ice condenser, and change the bases for the technical specifications.

Date of issuance: January 2, 1998.

Effective date: January 2, 1998, with full implementation within 45 days.

Amendment Nos.: 220 and 204.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54863).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085

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: October 15, 1997.

Brief description of amendment: Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1, 4.5.2, 4.6.2.1, and 4.6.2.2 require the recirculation spray, quench spray, residual heat removal, centrifugal charging, and safety injection pumps to be tested on a periodic basis and after modifications that alter subsystem flow characteristics. The amendment replaces the specific surveillance pump pressure with a statement that the test be conducted in accordance with Specification 4.0.5, Inservice Testing Program. The amendment also decreases the required individual safety injection and centrifugal charging pump injection line flow rates, increases the allowed individual safety injection pump runout flow rate, and makes editorial changes to the surveillances.

Date of issuance: December 24, 1997.

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

Amendment No.: 155.

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

Date of initial notice in **Federal Register***:* November 5, 1997 (62 FR 59918).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 24, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Public Service Electric & Gas Company, 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: November 4, 1997.

Brief description of amendments: These amendments revise Technical Specification 3/4.8.1 on the emergency diesel generators to (1) delete the 18-month surveillance requirements 4.8.1.1.2.d.1 and (2) eliminate the accelerated testing requirement of Table 4.8–1.

Date of issuance: January 8, 1998.

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

Amendment Nos.: 203 and 185.

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

Date of initial notice in **Federal Register***:* December 3, 1997 (62 FR 63982).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: November 21, 1997 (TS 97–05).

Brief description of amendments: The amendments change the Technical Specifications (TS) to allow a one-time provision for testing power-operated relief valves in Mode 5.

Date of issuance: January 13, 1998.

Effective date: January 13, 1998.

Amendment No.: 230.

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

Date of initial notice in **Federal Register***:* December 1, 1997 (62 FR 63565).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Dated at Rockville, Maryland, this 21st day of January 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–1904 Filed 1–27–98; 8:45 am]

BILLING CODE 7590–01–P

PEACE CORPS

Information Collection Requests Under OMB Review

ACTION: Notice of public use form review request to the Office of Management and Budget.

SUMMARY: Pursuant to the Paperwork Reduction Act of 1981 (44 USC, Chapter 35), the Peace Corps is requesting emergency approval and clearance from the Office of Management and Budget for use of the Peace Corps Day Brochure/Form to be used by the World Wise Schools program. A copy of the information collection may be obtained from Monica Fitzgerald, Office of World Wise Schools, Peace Corps, 1990 K St., NW, Washington, DC 20525. Ms. Fitzgerald may be called at (202) 606–9498. Peace Corps invites comments on whether the proposed collection of information is necessary for proper performance of the functions of the Peace Corps, including whether the information will have practical use; the accuracy of the agency's estimate of the burden of the proposed collection of information, including the validity of the methodology and assumptions used; ways to enhance the quality, utility and clarity of the information to be collected; and ways to minimize the burden of the collection of information on those who are to respond, including through the use of automated collection techniques, when appropriate, and other forms of information technology.

Comments on this form should be addressed to Victoria Becker Wassmer, Desk Officer, Office of Management and Budget, NEOB, Washington, DC 20503.

Information Collection Abstract

Title: Peace Corps Day Brochure/Form.

Need for and use of the Information: This form is completed voluntarily by Returned Peace Corps Volunteers and educators throughout the country. This information will be used by WWS to identify individuals interested in participating in the Peace Corps's annual Peace Corps Day program. Enrollment in this program also fulfills the third goal of Peace Corps as required by Congressional legislation and to enhance the Office of World Wise Schools global education program.

Respondents: Returned Peace Corps Volunteers and educators throughout the public and private school systems in the United States.

Respondents obligation to reply: Voluntary.

Burden on the Public:

a. Annual reporting burden: 4,750 hrs.

b. Annual record keeping burden: 0 hrs.

c. Estimated average burden per response: 3 min.

d. Frequency of response: annually.

e. Estimated number of likely respondents: 95,000.

f. Estimated cost to respondents: $0.79.

This notice is issued in Washington, DC on January 23, 1998.

Bessy Kong,

Acting Associate Director for Management.

[FR Doc. 98–2020 Filed 1–27–98; 8:45 am]

BILLING CODE 6051–01–M

PENSION BENEFIT GUARANTY CORPORATION

Submission of Information Collection for OMB Review; Comment Request; Allocating Unfunded Vested Benefits

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of request for extension of OMB approval.