Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the ''Final Environmental Statement Relating to the Operation of Duane Arnold Energy Center,'' dated March 1973.

Agencies and Persons Consulted

In accordance with its stated policy, the NRC staff consulted with the Iowa State official, Mr. D. Fleeter of the Department of Public Health, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the application dated June 14, 2000. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

Dated at Rockville, Maryland, this 3rd day of January 2001.

For the Nuclear Regulatory Commission.

John F. Stang,

Senior Project Manager, Section 1, Project Directorate III, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–731 Filed 1–9–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Public Meeting to Solicit Stakeholder Input on the Use of Risk Information in the Nuclear Materials Regulatory Process: Case Studies on Gas Chromatographs, Static Eliminators and Fixed Gauges

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) staff is developing an approach for using risk information in the nuclear materials and waste regulatory process. As part of this effort, the NRC staff is conducting case studies on a spectrum of activities in the nuclear materials and waste arenas, including the regulation of gas chromatographs, fixed gauges, and static eliminators. The purpose of the case studies is to illustrate what has been done and what could be done in the materials and waste arenas to alter the regulatory approach in a risk-informed manner, and to establish a framework for using a risk-informed approach in the materials and waste arenas by testing a set of draft screening criteria, and determining the feasibility of safety goals.

NRC staff is in the initial phase of the case studies on gas chromatographs, fixed gauges, and static eliminators. The purpose of this meeting is to: (1) Communicate to stakeholders the status of these case studies; (2) receive feedback and comments from stakeholders before continuing with the case studies; and (3) solicit from stakeholders comments or insights regarding the use of risk information in the NRC's regulation of gas chromatographs, fixed gauges, and static eliminators. The tentative agenda for the meeting is as follows:

1. Opening remarks.
2. Provide background information and general discussion on case studies.
3. Present status of case study on gas chromatographs and receive feedback and comments from meeting attendees.
4. Present status of case study on static eliminators and receive feedback and comments from meeting attendees.
5. Present status of case study on fixed gauges and receive feedback and comments from meeting attendees.
6. Receive general comments, feedback, and insights from meeting attendees with regard to the case studies and to using risk information in the NRC's regulation of gas chromatographs, fixed gauges, and static eliminators.
7. Closing remarks.

The meeting is open to the public; all interested parties may attend and provide comments. Persons who wish to attend the meeting should contact Marissa Bailey no later than January 29, 2001.

DATES: The meeting will be held on February 9, 2001, from 9 a.m. to 4 p.m., in the U.S. Nuclear Regulatory Commission Auditorium, 11545 Rockville Pike, Rockville, MD 20852.

FOR FURTHER INFORMATION CONTACT: Marissa Bailey, Mail Stop T–8–A–23, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Telephone: (301) 415–7648; Internet: MGB@NRC.GOV.

SUPPLEMENTARY INFORMATION: The NRC staff's case study approach, the draft screening criteria, and the case study areas under consideration are described in the ''Plan for Using Risk Information in the Materials and Waste Arenas: Case Studies'' which has been published in the **Federal Register** (65 FR 66782, November 7, 2000). Copies of this plan are also available on the Internet at *http://www.nrc.gov/NMSS/IMNS/riskassessment.html.* Written requests for single copies of this plan may also be submitted to the U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, Risk Task Group, Mail Stop T–8–A–23, Washington, DC 20555–0001.

Dated at Rockville, MD, this 4th day of January, 2001.

For the Nuclear Regulatory Commission.

Lawrence E. Kokajko,

Section Chief, Risk Task Group, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 01–732 Filed 1–9–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 18, 2000, through December 29, 2000. The last biweekly notice was published on December 27, 2000 (65 FR 81907).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By February 9, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* the Electronic Reading Room).

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

Date of amendment request: October 20, 2000.

Description of amendment request: The proposed amendment revises the Technical Specification (TS) surveillance interval for emergency diesel generator (EDG) maintenance from annually to 2 years. This interval is in conformance with guidelines of the Fairbanks Morris Owner's Group and the EDG manufacturer.

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

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

The changes do not affect the ability of the Emergency Diesel Generators (EDGs) to mitigate the consequences of an accident, including the loss of coolant accident coupled with loss of offsite power accident, which would be considered the most demanding on EDG System and components. A reduction in the number of diesel outages will also reduce the possibility of introducing problems resulting from human error or foreign material intrusion. Extending the maintenance interval should reduce the two-year unavailability from about 2% to about 1.4%. This is an approximate 30% reduction in unavailability. An extension of the outage inspection frequency to 24 months will result in increased EDG availability to mitigate the consequences of a potential accident. When this program is taken in its entirety, the extended maintenance intervals, coupled with the defined enhancements, is judged to result in an overall increase in Emergency Diesel Generator availability and reliability. The surveillance testing requirements of Technical Specifications Section 4.6.1a&b will continue to verify the operability and reliability of the EDG System. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The Emergency Diesel Generator System is not an accident initiator. The operation, testing, and design of the Emergency Power System (including the Emergency Diesel Generators) is not being changed. The maintenance inspection interval is being expanded from annual to two years and will improve availability and enhance reliability. Plant design requires the full load capability of one Emergency Diesel Generator to support accident loads and the respective emergency electrical busses. Performance of the maintenance inspection on the extended interval will not have an adverse affect on the ability of the Emergency Diesel Generators to meet the design response criteria or contribute to the occurrence or the consequences of an accident. The proposed changes do not involve any physical design or operation changes that could create a malfunction extending beyond an individual Emergency Diesel Generator, nor does it increase the potential for a common-mode Emergency Diesel Generator failure. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The change of the maintenance inspection frequency and the detailed programmatic changes that implement the Fairbanks Morse Owners Group recommendations, will increase the availability and reliability of the Emergency Diesel Generators. Based on improving the availability and reliability, the margin of safety will actually be enhanced. The amount of time the Emergency Diesel Generators are out-of-service during on-line maintenance will decrease, thereby reducing the number of plant operating hours that the unit is exposed to a single mode failure. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

Based on the analysis provided herein, the proposed change meets the requirements of 10 CFR 50.92(c) and involves no significant hazards consideration.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: September 14, 2000 as supplemented on December 21, 2000.

Description of amendment request: The proposed amendment incorporates the changes described below into the Technical Specifications (TSs) for Calvert Cliffs Unit 2. Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) also requested an exemption for Calvert Cliffs Unit 2 from the requirements of 10 CFR 50.46, 10 CFR 50.44, and 10 CFR Part 50, Appendix K.

The exemption and TS change will allow a lead fuel assembly (LFA) with a limited number of fuel rods clad with advanced zirconium-based alloys to be inserted into the core during the next Unit 2 refueling outage, scheduled to begin in March 2001. This LFA was approved to be inserted into Unit 1 Cycle 15. Because of concerns with corrosion performance, all of the Anikuloy, Alloy C, and Zr–2P clad rods were removed from this LFA and replaced with OPTIN and Alloy E rods from another LFA. Due to the length of time needed to perform this activity and the duration of the Unit 1 outage, it was not possible to reinsert this LFA into Unit 1 for Cycle 15 operation. Therefore, the licensee is requesting approval to insert this assembly into Unit 2 for Cycle 14 operation.

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

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

Supporting analyses indicate that since the LFA will be placed in a non-limiting location, the placement scheme and the similarity of the advanced alloys to zircaloy-4 will assure that the behavior of the fuel rods with these alloys are bounded by the fuel performance and safety analyses performed for the zircaloy-4 clad fuel rods currently in the Unit 2 Core. Therefore, the addition of these advanced claddings does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

The proposed change does not add any new equipment, modify any interfaces with existing equipment, change the equipment's function, or change the method of operating the equipment. The proposed change does not affect normal plant operations or configuration. Since the proposed change does not change the design, configuration, or operation, it could not become an accident initiator.

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

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

Supporting analyses indicate that since the LFA will be placed in a non-limiting location, the placement scheme and the similarity of the advanced alloys to zircaloy-4 will assure that the behavior of the fuel rods with these alloys are bounded by the fuel performance and safety analyses performed for the zircaloy-4 clad fuel rods currently in the Unit 2 Core. Therefore, the addition of these advanced claddings does not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Marsha Gamberoni.

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: October 20, 2000.

Description of amendment request: The amendments would revise the Technical Specification 3.7.10, ''Control Room Area Ventilation System (CRAVS).'' The primary purpose of the request is to eliminate the requirement for the CRAVS high chlorine protection function. Duke Energy Corporation (the licensee) indicated that a chlorine detection system with safety related detectors and automatic CRAVS intake isolation capability is no longer needed at Catawba. In addition, the licensee is also requesting NRC approval to allow the use of non-safety related detectors and to delete the automatic intake isolation capability. Finally, the amendments would also revise the Bases for the CRAVS to more clearly describe the system function and to make other clarifying 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:

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if 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.

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Neither the CRAVS, nor its automatic control room intake isolation function on a high chlorine condition is capable of initiating any accident. The CRAVS is responsible for maintaining an acceptable environment in the control room during normal operation and accident conditions. The CRAVS will continue to function as designed to provide this environment in accordance with all applicable TS. Following implementation of this amendment, Catawba plans to pursue elimination of the automatic intake isolation capability. This will not affect the system's ability to maintain an acceptable control room environment during and following an accident. No other design changes to the system are being made. It has been shown that the quantity of gaseous chlorine used at Catawba is less than the threshold stated in applicable Regulatory Guides. Hence, there is no control room habitability issue due to chlorine. Therefore, there will be no impact on any accident probabilities or consequences.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant which will introduce any new accident causal mechanisms. The elimination of the automatic intake isolation capability will not introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators and does not impact any safety analyses.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this amendment. The performance of the CRAVS in response to normal and accident conditions will not be impacted. There is no risk significance to this proposed amendment, as no reduction in system or component availability will be incurred. No safety margins will be impacted.

Based upon the preceding discussion, Duke has concluded that the proposed amendment does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: December 11, 2000.

Description of amendment request: The proposed amendment would modify the existing Minimum Critical

Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2. Specifically, the change modifies the MCPR Safety Limit value, as calculated by Global Nuclear Fuel, by increasing the limit for two recirculation loop operation from 1.09 to 1.10.

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.

Per the Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR) Section 4.2.1, the fuel system design bases are provided in the General Electric Standard Application for Reactor Fuel (GESTAR II). The Minimum Critical Power Ratio (MCPR) Safety Limit is one of the limits used to protect the fuel in accordance with the design basis. The NRC-approved MCPR Safety Limit calculations establish margin to the onset of transition boiling. The basis of the MCPR Safety Limit calculation remains the same, ensuring that greater than 99.9% of all fuel rods in the core avoid transition boiling. These NRC-approved calculations were used to determine the proposed limit, therefore there is not an increase in the probability of transition boiling. Also, the change does not result in any physical plant modifications or physically affect any plant components. Therefore, no individual precursors of an accident are affected. As a result, there is no increase in the probability of occurrence of a previously analyzed accident.

The fundamental sequences of accidents and transients have not been altered. The Safety Limit MCPR is established to avoid fuel damage in response to anticipated operational occurrences. Compliance with a MCPR safety limit greater than or equal to the calculated value will ensure that less than 0.1% of the fuel rods will experience boiling transition. This in turn ensures fuel damage does not occur following transients due to excessive thermal stresses on the fuel cladding. The MCPR Operating Limits are set higher (*i.e.,* more conservative) than the Safety Limit such that potentially limiting plant transients prevent the MCPR from decreasing below the MCPR Safety Limit during the transient. Therefore, there is no impact on any of the limiting USAR Appendix 15B transients. The radiological consequences remain the same as previously stated in the USAR. Therefore, the consequences of an accident do not increase over previous evaluations in the USAR.

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

The MCPR Safety Limit basis is preserved, which is to ensure that transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the limiting postulated transient. The value is calculated in accordance with GESTAR II. The GESTAR II analyses have been accepted by the NRC as comprehensive for ensuring that fuel designs will perform within acceptable bounds. The MCPR Safety Limit is one of the limits established to ensure the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. No changes in the design of the plant or the method of operating the plant are associated with this revised safety limit value. Therefore, no new accident precursors are created due to this change. As a result, no new or different kind of accident from any previously evaluated is created.

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

This change revised the PNPP MCPR Safety Limit value. The new MCPR Safety Limit value does not alter the design or function of any plant system, including the fuel. The new MCPR Safety Limit value was calculated using NRC-approved methods described in GESTAR II. The MCPR Safety Limit value is consistent with GESTAR II, the NRC Safety Evaluation of GESTAR II, and the Technical Specification Bases (Section 2.1.1.2) for the MCPR Safety Limit. Use of these methods satisfies the fuel design safety criteria that less than 0.1% of the fuel rods are predicted to experience transition boiling if the safety limit is not violated. Therefore, enforcing the new value for the MCPR Safety Limit does not involve a reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 23, 2000.

Description of amendment request: The proposed license amendments would revise Table 3.3–5, Accident Monitoring Instrumentation, and Table 4.3–4, Accident Monitoring Instrumentation Surveillance Requirements. The revision would delete reference to the containment hydrogen monitors from the Accident Monitoring Instrumentation. Additionally, the proposed amendments would delete Technical Specification (TS) 3/4.6.5, Combustible Gas Control—Hydrogen Monitors, and TS 3/4.6.6, Post Accident Containment Vent System.

In addition, the licensee requested an exemption from the requirements of 10 CFR 50.44, ''Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors and 10 CFR Part 50, Appendix E, Section VI, ''Emergency Response Data System.'' The purpose of the exemption is to remove the requirements for hydrogen-control systems from the Turkey Point (TP) Units 3 and 4 design basis. Moreover, the licensee's submittal requested a change to the Confirmatory Order dated March 14, 1983, and revised by NRC letter dated October 5, 2000, confirming TP Units 3 and 4 commitments related to NUREG–0737, post-TMI requirements. Specifically, the licensee requests deletion of the commitment to NUREG–0737, Item II.F.I, Item 6, Containment Hydrogen Monitor requirements. The exemption request and the revision to the Confirmatory Order will be evaluated separately from the proposed license amendments.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The Containment Combustible Gas Control System is composed of two hydrogen monitors, the Post-Accident Containment Vent System, and a leased hydrogen recombiner. Hydrogen control components are not considered to be accident initiators. Therefore, this change does not increase the probability of an accident previously evaluated.

The Containment Combustible Gas Control System is provided to ensure that the hydrogen concentration is maintained below 4.0% so that containment integrity is not challenged following a design basis Loss Of Coolant Accident (LOCA). Existing analysis show that the hydrogen concentration will not reach 4.0% for at least 12 days after a design basis LOCA. Containment failure due to hydrogen combustion without the Post-Accident Containment Vent System and backup hydrogen recombiner is not credible based on the results of the Turkey Point Units 3 and 4 Individual Plant Examination study. Therefore, this change does not increase the consequences of accidents previously evaluated.

Removal of the existing requirements for hydrogen control will reduce the consequences of postulated accidents by eliminating Post-Accident Containment Vent System releases, and by eliminating potential unfiltered release paths during operation of the hydrogen recombiner.

Removal of the existing requirements for hydrogen control will also allow elimination of the Emergency Operating Procedure (EOP) steps for hydrogen control and hence simplify migration through the EOPs. This would have a positive impact on public health risk by reducing the probability of operator error during potential accidents and hence reduce the core damage frequency. In addition, approval of these amendment requests will minimize the potential for actuation of the Post-Accident Containment Vent System and/or the backup hydrogen recombiner during severe accidents. The changes described in this request result in an overall decrease in risk.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This proposed change does not change the design or configuration of the plant beyond the containment Combustible Gas Control System. Hydrogen generation following a design basis LOCA has been evaluated in accordance with regulatory requirements. Deletion of the containment Combustible Gas Control System from the technical specifications does not alter the hydrogen generation processes post-LOCA. The consideration of hydrogen generation will no longer be included in the design basis of Turkey Point Units 3 and 4. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

No. The Containment Combustible Gas Control System is provided to ensure that the hydrogen concentration is maintained below 4.0% so that containment integrity is not challenged following a design basis Loss Of Coolant Accident (LOCA). Existing analysis show that the hydrogen concentration will not reach 4.0% for at least 12 days after a design basis LOCA. Containment failure due to hydrogen combustion without the Post-Accident Containment Vent System and backup hydrogen recombiner is not credible based on the results of the Turkey Point Units 3 and 4 Individual Plant Examination study. Therefore, this change does not result in a reduction in a margin of safety.

The changes proposed in these amendment requests result in a reduction in risk. Removal of the existing requirement for a containment Combustible Gas Control System will, by eliminating the EOP steps for hydrogen control, result in lower operator error probabilities. In addition, approval of these amendment requests will minimize the potential for actuation of the Post-Accident Containment Vent System and/or the backup hydrogen recombiner during severe accidents. Therefore, this change involves an increase in safety, not a reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

GPU Nuclear Corporation and Saxton Nuclear Experimental Corporation (SNEC), Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania

Date of amendment request: November 30, 2000.

Description of amendment request: The proposed amendment would change the name in the license of GPU Nuclear Corporation to GPU Nuclear, Inc.

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 NRC has previously determined that similar amendments reflecting this name change have involved no significant hazards consideration. See 62 Fed. Reg. 4341, 4350 (1997) and 62 Fed. Reg. 59912, 59915 (1997).

Consistent with these prior NRC determinations, GPU Nuclear has determined that the License Amendment involves no significant hazards considerations as defined in 10 CFR 50.92.

1. The proposed changes to the Saxton License do not involve a significant increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the safety analysis report. The changes have no impact on plant operations or the release of radioactive materials.

2. The proposed changes to the Saxton License will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report because no plant configuration or operation changes are involved.

3. The changes will not involve a significant reduction in the margin of safety as defined in the basis of any technical specification for Saxton because no change to operational limits will be made.

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

Attorney for the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Chief: Ledyard B. Marsh.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: December 5, 2000.

Description of amendment request: The proposed amendment would reformat the Technical Specifications to be more consistent with the proposed Improved Standard Technical Specifications applicable to permanently shutdown and defueled facilities. The proposed changes also modify the specifications to better reflect the decommissioned status of Millstone Nuclear Power Station, Unit 1. Other changes relocate requirements out of the Technical Specifications to other controlled license basis documents, consistent with the Improved Standard Technical Specifications and Nuclear Regulatory Commission (NRC) guidance.

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:

Administrative Changes (''A.x'' Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will not impose any new or eliminate any old requirements. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

Technical Changes—More Restrictive (''M.x'' Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, NNECO has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this 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 imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

''Generic'' Less Restrictive Changes: Relocating Details to Other Plant Controlled Documents (''LA.x'' Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, NNECO has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relocates certain details from the Technical Specifications to the TRM [Technical Requirements Manual] or the Millstone [Nuclear Power Station (Millstone),] Unit 1 Northeast Utilities Quality Assurance Program (NUQAP). The TRM will be maintained in accordance with 10 CFR 50.59. The NUQAP is subject to the change control provisions 10 CFR 50.54(a). Since any changes to the TRM or NUQAP will be evaluated per the requirements of 10 CFR 50.59 or 10 CFR 50.54(a) respectively, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the TRM, or the NUQAP documents are the same as the existing Technical Specifications. Since any future changes to these details in the TRM or NUQAP will be evaluated per the requirements of 10 CFR 50.59 or 10 CFR 50.54(a) respectively, no reduction (significant or insignificant) in a margin of safety will be allowed.

Relocated Specifications (''R.x'' Labeled Comments/Discussions

In accordance with the criteria set forth in 10 CFR 50.92, NNECO has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relocates requirements and surveillances for structures, systems, or components (SSCs) that do not meet the criteria for inclusion in Technical Specifications as defined in 10 CFR 50.36. The affected SSCs are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected SSCs will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected SSCs are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant activities. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected SSCs remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

Specific Less Restrictive Changes (L.1 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. This change modifies the Applicability of LCO 3.1.1 from ''Whenever irradiated fuel is stored in the Fuel Storage Pool'' to ''During movement of irradiated fuel assemblies in the Fuel Storage Pool.'' This is consistent with the conditions addressed and assumed in the analysis of a fuel handling accident. Required Action A.2 is also deleted since, with the corresponding change to the Applicability, it is no longer required. The following is provided in support of this conclusion.

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

The proposed change involves modifying the Applicability of LCO 3.1.1 to correspond directly with the conditions to which the LCO applies. LCO 3.1.1 provides assurance that adequate pool water level is maintained to ensure that the assumptions of the design basis fuel handling accident are met. The design basis accident assumes a non-mechanistic failure of the fuel pins in four assemblies. The analysis assumes that a water level below that required by LCO 3.1.1.

If fuel handling is not occurring, the fuel pool water level does not satisfy the criteria for inclusion in the Technical Specifications as a parameter assumed as an initial condition of the safety analysis. Therefore this change merely aligns the LCO Applicability with the safety analysis assumptions.

Aligning the Applicability directly with the conditions that must exist for a design basis accident to occur does not affect the probability or consequences of an accident previously evaluated. Rather, it ensures that the previously evaluated accident probability and consequences are unchanged. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will merely align the Applicability of an existing LCO with the conditions that exist when the limit of the LCO is credited in the safety analysis. Thus, this 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 will not reduce a margin of safety because the change merely aligns the Applicability of LCO 3.1.1 with the conditions that exist when the limit of the LCO is credited in the safety analysis. Therefore, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.2 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The proposed change removes a restriction from Section 4.1, Site Location, which restricts the sale or lease of portions of the site other than to the listed organizations. The following is provided in support of this conclusion.

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

The proposed change involves removing an administrative restriction on the ownership and ability to lease portions of the site to organizations other than those listed. Removing this restriction will not affect the probability of an accident previously evaluated, since these restrictions are not related to any precursor or contributor to the causes for any accident previously evaluated. Removing the restrictions will similarly not increase the consequences of an accident previously evaluated, since the proposed change does not result in a transfer of ownership or grant of lease of the described property. Any such activity would be subjected to a review in accordance with the requirements of 10 CFR 50.59, since the ownership and physical description of the plant are described in the Defueled Safety Analysis Report. The evaluation performed at that time would ensure that no increase in the consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change merely removes an administrative requirement that limits the ability to sell or lease portions of the site. These controls are not associated with any onsite activity that could result in a new or different kind of accident. Thus, this 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 will not reduce a margin of safety, because it does not result in any change to the plant or the way it is operated. The proposed change merely removes an administrative restriction on the ability to lease or sell portions of the site. Since the site description is provided in the Defueled Safety Analysis Report, any such activity would be subject to a review in accordance with the requirements of 10 CFR 50.59. This review would ensure that there is no reduction in margin of safety associated with any future proposed changes. Therefore, this change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.3 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The proposed change removes a limit associated with the storage of fuel in the new fuel storage facility. With the permanent shutdown and defueled condition of the plant, and the removal of all un-irradiated fuel from the site, the new fuel storage facility will no longer be used and this restriction is no longer required. The following is provided in support of this conclusion.

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

The proposed change involves removing restrictions on k_{eff} in the new fuel storage facility. Fuel can no longer be stored in the new fuel storage facility because all un-irradiated fuel has been removed from the site, and radiological considerations prevent the placement of irradiated fuel in the new fuel storage facility. The design basis accident for Millstone, Unit No. 1 is the postulated Fuel Handling Accident described in the Defueled Safety Analysis Report. The postulated accident involves irradiated fuel located in the spent fuel storage pool. Therefore, this requirement provides no useful information and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will not impose any new requirements. Thus, this 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 will not reduce a margin of safety, because the requirements that are proposed for elimination do not affect the design or operation of the facility since the plant was permanently shutdown, defueled, and all un-irradiated fuel has been removed from the unit. Since the proposed change has no affect on the facility and merely removes unnecessary information from the Technical Specifications, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.4 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The proposed change involves removing the requirement for a Shift Manager who is qualified as a Certified Fuel Handler and is responsible for the control room command function. The following is provided in support of this conclusion.

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

The proposed change involves removing the requirement for a Shift Manager who is qualified as a Certified Fuel Handler and who is responsible for the control room command function. Millstone, Unit No. 1 has been shutdown for over four years, and there are no remaining postulated or credible accidents that require a complex immediate response from operating personnel. The required response to postulated and credible accidents at the facility are a small subset of those that were required when the facility was in operation. Based on this, there is no longer a need for a specific position designation for the individual who will exercise the control room command function.

In addition, the requirement for a Certified Fuel Handler to fulfill the Shift Manager responsibility is no longer appropriate because for extended periods no fuel handling operations will be conducted. Fuel Handling activities are deliberate pre-planned evolutions. There are no postulated or credible accidents that would result in the need to perform an unplanned fuel movement. Plant procedures and other administrative controls will continue to ensure that Certified Fuel Handler responsibilities are fulfilled by appropriately qualified individuals when activities dictate the need.

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities because qualified individuals will continue to be available to perform required functions. The proposed change will not impose any new or eliminate any old requirements associated with any structure, system or component. Thus, this 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 will not reduce a margin of safety, because qualified individuals will continue to be available to perform activities required to ensure the safe storage of irradiated fuel and control of radioactive materials. The proposed changes will eliminate unnecessarily burdensome requirements that were developed to address the requirements of an operating facility but which no longer apply at a permanently shutdown and defueled facility such as Millstone, Unit No. 1. Therefore, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.5 Labeled Comments/Discussions)

DOC L.5 is not used.

Specific Less Restrictive Changes (L.6 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The proposed change removes an administrative requirement for notification to be made to the NRC prior to changes to acceptance criteria for chemistry control of the Fuel Storage Pool. The following is provided in support of this conclusion.

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

Removing the requirement for prior notification of the NRC cannot have any effect on the probability or consequence of an accident previously evaluated, since the requirement to perform this notification is not associated or related in any way to the probability or consequences of any accident.

The consequence of an accident previously evaluated are not affected since no change to the way the fuel storage pool is monitored, is proposed. Notification of the NRC does not affect the consequences of any previously evaluated accident. The proposed change merely reduces the administrative burden associated with maintaining the program in compliance with the Technical Specifications.

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

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed changes will not impose any new or eliminate any old requirements. Thus, these changes do 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 changes will not reduce a margin of safety because they merely remove administrative burden associated with implementing the Fuel Storage Pool Program by eliminating a requirement for notification to the NRC of proposed changes to acceptance criteria to be used. Therefore, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.7 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The proposed change merely adds the option to use electronic dosimetry. The following is provided in support of this conclusion.

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

The proposed change involves adding the explicit option to utilize electronic dosimetry as a means of monitoring occupational radiation exposure. The means of monitoring occupational dose are unrelated to the probability or consequences of any accident previously evaluated. The means of measuring occupational exposures is merely a limit on the technology that may be utilized to perform a measurement required by [F]ederal regulations. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of plant systems, structures or components (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will not impose any new or eliminate any old requirements related to the safe storage of irradiated nuclear fuel or the control of radioactive materials. Thus, this 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 will not reduce a margin of safety, because the means of measuring the occupational exposure of workers is unrelated to the margin of safety of the facility. The means of measuring occupational exposures is merely a limit on the technology that may be utilized to perform a measurement required by [F]ederal regulations.

Specific Less Restrictive Changes (L.8 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. This change will extend the surveillance Frequency from once every 24 hours to once every [seven] days. The proposed Frequency is consistent with the reduced decay heat load and the lack of available mechanistic failures that could lead to sudden or unanticipated reduction in spent fuel pool inventory. The associated Bases are modified to reflect the proposed interval. The following is provided in support of this conclusion.

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

The proposed change involves extending the Frequency interval of SR [Surveillance Requirement] 3.1.1 to correspond with the conditions of the facility. SR 3.1.1 provides assurance that adequate pool water level is maintained to ensure that the assumptions of the design basis fuel handling accident are met. There are no longer any credible mechanisms that could lead to an unanticipated or undetected reduction in spent fuel pool inventory. The proposed [seven] day Frequency is consistent with the decay heat load calculations, potential maximum evaporation rates, and the large volume of water available over the spent fuel in the storage pool.

Aligning this SR directly with the conditions that exist in the facility does not affect the probability or consequences of an accident previously evaluated. Rather, it continues to ensure that the previously evaluated accident probability and consequences are unchanged. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will merely align the Frequency of an existing SR with the conditions in the facility. Thus, this 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 will not reduce a margin of safety because the change merely aligns the Frequency of performance of SR 3.1.1 with the conditions that exist in the plant. Therefore, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.9 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. This change modifies the spent fuel storage rack limit on K_{eff} from less than or equal to 0.90 to less than or equal to 0.95. The following is provided in support of this conclusion.

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

The proposed change involves modifying the k_{eff} limit that the spent fuel storage racks are designed and maintained to. The current and proposed limit are established to provide a significant margin of assurance that the spent fuel cannot be made critical while stored in the racks and under design basis accident conditions.

Changing the limit on k_{eff} from 0.90 to 0.95 does not significantly affect the assurance that the spent fuel racks will maintain the fuel in a sub-critical configuration. Both limits are substantially below the limit of 1.0, and provide adequate assurance of safety. The proposed change therefore does not affect the probability or consequences of an accident previously evaluated. Rather, it continues to ensure that the previously evaluated accident probability and consequences are unchanged. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will merely increase the limit on k_{eff} so that it is consistent with industry practice and established standards applicable to the storage of spent fuel. Criticality continues to be avoided by maintaining the storage racks such that k_{eff} is less than or equal to 0.95. Thus, this 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 margin of safety defined by the limit is that the spent fuel will remain sub-critical during anticipated circumstances and design basis accidents. Since the proposed limit continues to provide this assurance, the change does not involve a significant reduction in a margin of safety.

Specific Less Restrictive Changes (L.10 Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. This change removes the redundant requirement to maintain an NRC approved training and retraining program for the Certified Fuel Handlers (CFHs). The following is provided in support of this conclusion.

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

The proposed change removes a TS [Technical Specification] administrative requirement that is redundant to existing requirements that derive from 10 CFR 50.2. Therefore the TS requirement is not needed and does not [a]ffect the probability or consequences of an accident previously evaluated. The change is purely administrative, albeit a specific reduction in the requirements of the TS. The requirement will continue to apply to the unit. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant activities. The proposed change will merely remove an unneeded, redundant requirement. Therefore, this 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 will not reduce a margin of safety because the requirement for an NRC approved training program for CFHs will continue to exist as specified in 10 CFR 50.2. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Michael T. Masnik.

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

Date of amendment request: August 3, 2000.

Description of amendment request: The proposed amendment would delete Section 3.D, ''License Term'' from Facility Operating License No. DPR–40. The long-term load factor described in Section 3.D is used in the projection of reactor vessel fast neutron fluence and consequently for calculation of the pressurized thermal shock (PTS) reference temperature (RT_{PTS}) value to ensure that the 10 CFR 50.61 screening criteria for reactor vessel integrity are not exceeded. The previous fluence analysis was performed by Combustion Engineering (ABB/CE). Recently, Westinghouse Electric Company has completed an analysis (WCAP–15443, ''Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel,'' dated July 2000) to update the ABB/CE calculation. In accordance with 10 CFR 50.61, this assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon request for a change in the expiration date of the facility. Thus, Section 3.D can be deleted from Facility Operating License No. DPR–40 based upon the recent Westinghouse analysis and the fact that Section 3.D is redundant to 10 CFR 50.61 requirements.

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 previously evaluated accidents affected by this change are limited to the pressurized thermal shock (PTS) events. Vessel embrittlement due to fast neutron associated damage to the limiting beltline region reactor vessel material (which for Fort Calhoun Station is included in the lower course axial welds) is a component in the PTS analysis. The fast neutron, thermal neutron and dpa [displacement per atom] values of the FCS reactor vessel were recalculated using actual power history values for Cycles 1 through 14 rather than conservative estimates, along with the revised BUGLE–93 cross sections from the ENDF/B–VI cross section library to appropriately account for the iron atoms in the thermal shield and a methodology that the NRC has previously approved for neutron fluence calculations performed by Westinghouse. The fluence evaluation included data from the three surveillance capsules (W–225, W–265, and W–275) previously removed and analyzed. The RT_{PTS} evaluation applied Position 2.1 of Regulatory Guide 1.99, Revision 2 in conjunction with surveillance data from other plants containing the limiting FCS weld materials. The evaluation results indicate that the FCS reactor vessel is able to reach more than 20 years beyond current licensed life without exceeding the 10 CFR 50.61 screening criterion for RT_{PTS} of 270°F for axial welds.

In accordance with 10 CFR 50.61, this assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon request for a change in the expiration date of the facility. Since these requirements are contained in 10 CFR 50.61, Section 3.D can be deleted from Operating License No. DPR–40 without resulting in 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 does not physically alter the configuration of the plant and no new or different mode of operation is proposed. Increasing the long term load factor from 0.77 to 0.85 more accurately projects RT_{PTS} by accounting for improvement in FCS operating cycle efficiency. Requirements for assessing and reporting RT_{PTS} are contained in 10 CFR 50.61 and therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The margin of safety is defined by both the screening criteria of 10 CFR 50.61 and draft regulatory guide DG–1053 for neutron fluence calculations, which requires the methodology to be capable of providing best estimate fluence evaluations within 20 percent (1σ). The analysis for FCS shows that when the applicable regulatory criteria are applied, the screening criteria of 10 CFR 50.61 are not exceeded; therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 14, 2000 (This supercedes the license amendment request dated July 28, 2000, that was published in the **Federal Register** on October 18, 2000 (65 FR 62388).

Description of amendment request: The proposed amendment would permit Fort Calhoun Station to install leak tight sleeves as an alternative to plugging to repair defective steam generator tubes.

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

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

The CE Leak Tight Sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applicable design criteria for the sleeves conform to the stress limits and margins of safety of Section III of the ASME code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

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

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

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

In conclusion, based on the discussion above, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

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

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. As discussed above, the reduced primary to secondary leakage limit is a conservative change in the plant limiting conditions for operation. Therefore, Omaha Public Power District concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 5, 2000.

Description of amendment request: The proposed change revises Surveillance Requirement 4.6.3.4 to require testing of a representative sample of Excess Flow Check Valves (EFCVs) such that each EFCV will be tested at least once every 120 months.

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

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

The current Surveillance Requirement (SR) frequency requires each reactor instrumentation line Excess Flow Check Valve (EFCV) to be tested every 24 months. The EFCVs at LGS (Limerick Generating Station), Units 1 and 2 are designed to not close accidentally during normal operation, but will close automatically in the event of a line break downstream of the valve. The EFCVs are provided with valve position indication in the reactor enclosure. A general alarm is provided in the control room to indicate that an EFCV position has changed state. As discussed in the LGS, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) (Section 6.2.4.3.1.5), instrument lines that penetrate the containment from the Reactor Coolant Pressure Boundary (RCPB) conform to Regulatory Guide 1.11 in that they are equipped with a restricting orifice located inside the drywell and an EFCV located outside the drywell as close as practical to the containment. The GE Nuclear Energy (GENE) Report demonstrates, through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. A failure of an EFCV to isolate cannot initiate previously evaluated accidents. In addition, since the proposed changes will only change the surveillance frequency, there can be no increase in the probability of occurrence of an accident as a result of this proposed change.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Section 15.6.2. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are substantially below the guidelines of 10 CFR 100. Therefore, a failure of an EFCV, though not expected as a result of the change in the surveillance frequency, is bounded by the previous evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by this proposed change. Therefore, the proposed TS changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes allow a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in the GENE report provides supporting evidence that the reduced testing frequency does not affect the kind of accident. The potential failure of an EFCV to isolate as a result of the proposed reduction in test frequency is not a physical alteration of the plant and will not alter the operation of the structures, systems and components as described in the UFSAR. Therefore, a new or different kind of accident will not be created.

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

The consequences of an unisolable rupture of an instrument line has been previously evaluated in the LGS, Units 1 and 2 UFSAR, Section 15.6.2. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence. The change in surveillance frequency only changes the potential for an undetected failure of an EFCV and does not change the event sequence upon which the current margin is based. Therefore, no change in the margin of safety.

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 4, 2000.

Description of amendment request: The amendment proposes changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TSs) to revise the surveillance requirement (SR) for certain isolation valves known as excess flow check valves (EFCV). The current TSs require that each EFCV be tested at least once every 24 months. The proposed change would allow a representative sample to be tested every 24 months, such that each EFCV is tested at least once every 10 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:

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

This proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

The current SR frequency requires each reactor instrumentation line EFCV to be tested every 24 months. The EFCVs at SSES Unit 1 and Unit 2 are designed so that they will not close accidentally during normal operation, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and have their status indicated in the control room. This proposed change allows a reduced number of EFCVs to be tested every 24 months. There are no physical plant modifications associated with this change. Industry and SSES operating experience demonstrates a high reliability of these valves. Neither EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore there can be no increase in the probability of occurrence of an accident regarding this proposed change.

The SSES FSAR [Final Safety Analysis Report] Section 15.6.2 demonstrates (consistent with the BWROG [Boiling Water Reactor Owners' Group] report) that the failure of an EFCV has very low consequence. SSES FSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation assumes the EFCV fails to isolate the break. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10 CFR 100 limits. Thus the failure of an EFCV, though not expected as a result of this proposed change, does not affect the dose consequences of an instrument line break.

Based on the above, it is concluded that the proposed change to the EFCV surveillance requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry and Susquehanna-specific operating experience demonstrates the high reliability of these valves. The potential failure of an EFCV to isolate by the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line rupture. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created from implementation of the proposed change.

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

SSES FSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation assumes the EFCV fails to isolate the break. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10CFR100 limits. Thus the failure of an EFCV, though not expected as a result of this proposed change, does not affect the dose consequences of an instrument line break.

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

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: November 16, 2000.

Description of amendment request: The request for amendment proposes changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications to eliminate response time testing requirements for certain reactor protection system and isolation actuation system instrumentation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, 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 proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change eliminates certain response time testing [RTT] surveillance requirements in accordance with the NRC [Nuclear Regulatory Commission] approved methodology delineated in the BWROG [Boiling Water Reactor Owners' Group] Licensing Topical Report [LTR] NEDO 32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' dated October 1995, and its Supplement 1, dated October, 1999.

Implementation of the LTR and its supplement (*i.e.,* elimination of response time testing for selected instrumentation in the Reactor Protection System [RPS] and Isolation Actuation System [IAS]) does not increase the probability or consequences of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR [Final Safety Analysis Report]. All component models used in the affected trip channels at SSES were analyzed for a sluggish response, or a bounding response time. As documented in the LTR and supplement, the component's sluggish response can be detected by other Technical Specification required tests. The bounding response time of the relays discussed in the LTR Supplement 1 can be used in place of actual measured response times to ensure that instrumentation systems will meet response time requirements of the accident analysis. Response Time Testing for the channel process sensors are also eliminated on a similar basis, or have previously been eliminated in license amendments (171 (Unit 1) and 144 (Unit 2)).

Based upon the analysis presented above, PPL concludes that the proposed action does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change eliminates certain response time testing (RTT) surveillance requirements in accordance with the NRC approved methodology delineated in the BWROG Licensing Topical Report (LTR) NEDO 32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' dated October 1995, and its Supplement 1, dated October, 1999.

Implementation of the LTR methodology and the Supplement methodology does not create the probability of a new or different type of accident from any accident previously evaluated. A review of the failure modes of the affected sensors and relays indicates that a sluggish response of the instruments can be detected by other Technical Specification surveillances. A review of SSES RTT history (in support of the LTR) revealed one RTT failure. This failure would have been detectable by the logic system functional test for this channel. Redundancy and diversity of the affected channels provide additional assurance that all affected functions will operate within the acceptance limits of the safety evaluations.

The sensors and relays in the affected RPS and IAS channels will be able to meet the bounding response times as defined and presented in the Supplement. It has been found acceptable to use component bounding response times in place of actual measured response times to ensure that instrumentation systems will meet response time requirements of the accident analysis.

PPL's adherence to the conditions listed in the NRC SERs [Safety Evaluation Reports] for the LTR and Supplement provides additional assurance that the instrumentation systems will meet the response time requirements of the accident analyses.

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

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

The change does not involve a significant reduction in the margin of safety. The proposed change eliminates certain response time testing (RTT) surveillance requirements in accordance with the NRC approved methodology delineated in the BWROG Licensing Topical Report (LTR) NEDO 32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' dated October 1995, and its Supplement 1, dated October, 1999.

Implementation of the LTR and Supplement methodologies for eliminating selected response time testing does not involve a significant reduction in the margin of safety. The current response time limits are based on the maximum allowable values assumed in the plant safety analyses. The analyses conservatively establish the margin of safety. The elimination of the selected response time testing does not affect the capability of the associated systems to perform their intended function within the allowed response time used as tile basis for plant safety analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore, the margin of safety is not affected. This is based upon the ability to detect a sluggish response of an instrument or relay by the other required Technical Specification tests, component reliability, and redundancy and diversity of the affected functions, as justified in the reviewed and approved Topical Report and Supplement.

PPL's adherence to the conditions listed in the NRC SERs for the LTR and Supplement provides additional assurance that the instrumentation systems will meet the response time requirements of the accident analyses.

Thus, PPL concludes that the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: November 28, 2000.

Description of amendment request: The proposed change would modify Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.6.3 to expand the allowable vacuum breaker open differential pressure setpoint range to ≥ .25 pounds-per-square-inch differential (psid) and ≥ .75 psid. The SR in the current TSs requires testing to a range of ≥ .25 psid and ≥ .525 psid.

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.

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

The SR is required to verify that the vacuum breakers open when required by the containment safety analysis. The vacuum breakers setpoint prevent the creation of a vacuum in the drywell or an unacceptable differential pressure across the containment diaphragm slab. When the drywell pressure falls below the airspace pressure by an amount equal to the open set pressure of the vacuum breakers, the vacuum breakers open to allow the suppression chamber atmosphere from the wetwell airspace to flow into the drywell.

The ability to maintain containment integrity is not affected by the proposed change. Containment analyses are not affected by the proposed change.

Containment analyses assume the vacuum breakers open at .9 psid. Thus, the vacuum breakers at the new setpoint range are bounded by the setpoint assumed in the analysis. Sensitivity analyses show that the containment pressure response is insignificantly affected by the proposed change.

The setpoint expansion does not adversely affect the vacuum breakers ability to perform their design basis functions.

Based on the above, it is concluded that the proposed change to the vacuum breakers setpoint surveillance requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change allows an expanded setpoint range. No other changes in requirements are being proposed. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. The new range is bounded by the safety analysis assumptions. Thus, a new or different kind of accident will not be created from implementation of the proposed change.

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

This proposal does not involve a significant reduction in the margin of safety.

The containment pressures are insignificantly affected by the proposed change. Safety analyses assume a bounding setpoint. The operation of the vacuum breakers are not affected.

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

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: August 17, 2000.

Description of amendment request: The proposed amendments would eliminate the need for the licensee to perform periodic response time testing of selected reactor trip system and engineered safety feature actuation system equipment as defined in Westinghouse report WCAP–14036–P–A Revision 1, ''Elimination of Periodic Protection Channel Response Time Tests.''

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:

Conformance of the proposed amendment to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 is shown in the following.

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

This change to the Technical Specifications does not result in a condition where the design, material, and constructionstandards that were applicable prior to the change are altered. The same RTS [Reactor Trip System] and ESFAS [Engineered Safety Features Actuation] instrumentation is being used. The time response allocations and modeling assumptions used in the Chapter 6 and Chapter 15 safety analyses of the Final Safety Analysis Report (FSAR) are not changed; only the method of verifying response time is changed. The proposed change will not modify any system interface or equipment design specification. The proposed change can not increase the likelihood of an accident since such postulated events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR. Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not alter the performance of the protection channel and actuation logic equipment used in the RTS and ESFAS. These protection systems will still have response time verified by test before being placed in operational service. Changing the method of periodically verifying instrument response for these systems (assuring equipment operability) from time response testing to calibration and functional testing will not create any new accident initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristic to exceed the total allowance. The total time response allowance for each function and the response time allowance for individual components (e.g., circuit boards and relays) bound all degradation that cannot be detected by periodic surveillance. Therefore, implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total RTS and ESFAS response times assumed in the safety analyses. The periodic response time verification method for the 7300 Process Protection racks, NIS [Nuclear Instrumentation System] racks, and SSPS [Solid-State Protection System] actuation logic is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis. Periodic calibrations and functional tests will continue to be performed and may be used to detect degradation which might cause the response time to exceed the total allowance. The time response allowance for each component and function bounds all degradation that cannot be detected by periodic surveillance. Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Richard L. Emch, Jr..

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

Date of amendment request: December 6, 2000.

Brief description of amendment request: The proposed license amendment request will revise Administrative Controls in Technical Specification (TS) 5.5.14, entitled ''Technical Specifications (TS) Bases Control Program'' and TS 5.5.17, entitled ''Technical Requirements Manual (TRM)'' to incorporate the changes made to 10 CFR 50.59 as published in the October 4, 1999, **Federal Register**, Volume 64, Number 191, ''Changes, Tests, and Experiments,'' pages 53582 through 53617.

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

Response: No.

The proposed changes replace the word ''involve(s)'' with ''require(s)'' and deletes reference to the term ''unreviewed safety question.'' The above changes are consistent with the revision to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not increased. Changes to the Technical Specification (TS) Bases and the Technical Requirements Manual (TRM) are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not affected.

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

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

Response: No.

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. These changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements in the TS.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

The proposed changes will not reduce the margin of safety because they have no direct effect on any safety analyses assumptions. Changes to the TS Bases and the TRM that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval. The proposed changes to TS 5.5.14 and TS 5.5.17 are considered administrative in nature based on the revisions to 10 CFR 50.59.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 27, 2000.

Description of amendment request: This proposed change eliminates the specifications associated with the 24 Vdc Emergency Core Cooling System (ECCS) instrumentation batteries and chargers. The 24 Vdc ECCS instrumentation loads will be transferred to the 125 Vdc main station batteries.

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

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

The loads previously supplied by the ECCS battery systems will be added to the main station battery systems. Redundancy and reliability are maintained within the main station battery systems and the equipment will operate, essentially the same. No change in accident assumptions or pre[]cursors are involved with this change and system operation and response to analyzed events is likewise unchanged.

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

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

The methods by which the DC system supplied equipment performs their safety functions are unchanged and remain consistent with current safety analysis assumptions. The redundancy and reliability of the equipment will be maintained. There is no change in system or plant operation that involves failure modes other than those previously evaluated.

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

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

No adverse effect on equipment operation, capability or reliability will result from this change. The equipment supplied by the DC systems involved in this change will continue to be provided with adequate, redundant, reliable, safety class DC power. Safety related loads will continue to function in accordance with analysis assumptions.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 30, 2000.

Description of amendment request: The proposed change would revise the operability requirements for the refueling interlocks contained within Technical Specification (TS) 3.12.A as well as the surveillance requirements specified within TS 4.12.A. In addition, TS 3.12.F will be clarified to articulate that there must be a minimum of 24 hours fission product decay prior to fuel handling.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The only accident described within the [Final Safety Analysis Report] FSAR while the plant is in Cold Shutdown or Refueling is a fuel handling (dropped bundle) accident. The proposed change involves equipment that is not involved in the mitigation or prevention of a fuel handling accident as described in the FSAR. Accordingly, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will not effect the ability of the refueling interlocks to satisfy the safety function which is to prevent reactor criticality during refueling operations. The change only effects those interlocks which are not instrumental in satisfying the safety function of the interlocks.

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

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

The proposed change does not involve any physical alteration of plant equipment or to the status of the reactor core during refueling. The specifications will ensure either through the interlocks or the proposed alternative, that control rods are not withdrawn and cannot be inappropriately withdrawn. This will ensure that fuel is not loaded into the core when a control rod is withdrawn.

Therefore, no new failure modes are introduced and the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not involve a significant reduction in a margin of safety since the refueling interlocks will continue to ensure against an inadvertent criticality. This is achieved by physical interlocks or Technical Specification restrictions on refueling operations which will prevent fuel from being loaded into a core cell void of a control rod. This is accomplished by blocking control rod withdrawal whenever fuel is being loaded into the reactor vessel or by preventing fuel from being loaded into the vessel when a control rod is withdrawn.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 19, 2000. This supersedes the March 17, 2000, submittal which was noticed on May 3, 2000 (65 FR 25769).

Description of amendment request: The proposed changes would modify the voltage setting limits specified in Technical Specification (TS) Table 3.7–4, page 3.7–26, item 7 for the emergency bus degraded voltage, and revise the loss of voltage setpoints from a percentage of nominal bus voltage to an actual bus voltage value. The degraded voltage setting limit is being changed to increase the minimum allowable bus voltage to improve long-term motor performance in the event of operation with bus voltage less than nominal. The emergency bus loss of voltage setting limit is being revised to better address expected relay performance over time (*i.e.,* setting drift). Section 3.6.B, page 3.6–1, of the TS would be changed to revise the required reactor coolant system conditions from the existing wording of ''350 degrees F or 450 psig'' to ''350 degrees F and 450 psig.''

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:

We have reviewed the proposed change against the criteria of 10 CFR 50.92 and have concluded that the change does not pose a significant safety hazards consideration as defined therein. Specifically, operation of Surry Power Station with the proposed change will not:

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

No increase in the probability of occurrence or consequences of an accident previously evaluated will result from the proposed change in the setting limits for the emergency bus degraded voltage and loss of voltage relay setpoints. The proposed change only affects actuation limits and therefore has no bearing on the probability of an accident. Neither the logic nor the function of the undervoltage protection circuits is being changed, nor is circuit or equipment reliability being reduced. Further, the performance characteristics of the electrical distribution system and components supplied (motors, etc.) are not being altered, and compliance with GDC–17 [General Design Criterion] is being maintained. The electrical distribution system remains capable of performing its safety function without spurious separation of the emergency buses from offsite power. If offsite power is lost, the capability of the EDG's [emergency diesel generators] to perform their safety function is not altered. Therefore, the probability of an accident previously evaluated is not increased.

The consequences of an accident would not increase since the proposed change implements setting limits that will continue to ensure that adequate voltages will be available for the continuous operation of safety-related equipment required to function to mitigate a design basis accident. The proposed setting limits for the emergency bus degraded voltage and loss of voltage bound the setpoints and initial conditions assumed in the accident analyses and ensure that appropriate protection is maintained.

The editorial change is administrative in nature and consequently does not affect the probability or consequences of an accident in any way.

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

Implementing the proposed Technical Specifications emergency bus degraded voltage and loss of voltage relay setting limits cannot create the possibility of a new or different kind of accident than any accident previously evaluated. Revising the setpoint setting limits does not introduce any new accident precursors, and operation of the electrical distribution system and the undervoltage relaying scheme is unchanged. The relays will continue to detect undervoltage conditions and transfer safety loads to the emergency diesel generators at a voltage level adequate to ensure proper safety equipment performance and to prevent long-term equipment degradation due to undervoltage conditions. The proposed setting limits include adequate tolerances to calibrate the undervoltage relays while ensuring that emergency bus voltages remain above analytical limits. As noted above, the performance characteristics of the electrical distribution system and the components being supplied are not being altered, and compliance with GDC–17 is being maintained.

The editorial change is administrative in nature and consequently does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change continues to ensure that adequate voltage is available for safety-related equipment relied upon to respond to a design basis accident. The proposed setting limit for degraded bus voltage is conservative with respect to the existing Technical Specifications and ensures an adequate safety margin is being maintained. Further, the setting limit is maintained low enough to prevent spurious actuations given expected offsite grid voltages. While the loss of bus voltage setting limit is being expanded, sustained bus voltage in this range is not credible. Furthermore, there is no safety limit associated with the loss of voltage setting limit. The proposed change continues to ensure that the setting limits for the emergency bus degraded voltage and loss of voltage relays bound the setpoints and initial conditions assumed in the accident analyses and ensures that appropriate electrical protection is maintained. The editorial change is administrative in nature and consequently does not affect the safety analysis in any way. Consequently, the margin of safety is not being reduced by the proposed Technical Specifications change.

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

Attorney for licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: December 7, 2000 (ET 00–0044).

Description of amendment request: The proposed amendment adds text to Section 5.5.2, ''Primary Coolant Sources Outside Containment,'' and deletes Section 5.5.3, ''Post Accident sampling,'' from the administrative controls section of the Technical Specifications (TS). The proposed amendment deletes requirements from the TS (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in a license amendment application in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated December 7, 2000.

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

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

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

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

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

Therefore, the elimination of PASS requirements from Technical Specifications

(TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

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

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

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

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

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 8, 2000.

Description of amendment request: The proposed amendment would revise Administrative Controls Technical Specifications (TS) 5.5.14b and 5.5.14b.2 to incorporate the changes made to 10 CFR 50.59. The proposed changes would replace the word ''involve'' with ''require'' in TS 5.5.14b and revise TS 5.5.14b.2 to state: ''a change to the USAR [Updated Safety Analysis Report] or Bases that requires NRC approval pursuant to 10 CFR 50.59.''

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

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

The proposed changes replace the word ''involve'' with ''require'' and deletes reference to the term ''unreviewed safety question'' consistent with 10 CFR 50.59. The above changes are consistent with the revision to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not increased. Changes to the Technical Specification (TS) Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not affected.

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

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. These changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements in the TS.

Therefore, the proposed changes do not create 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 will not reduce the margin of safety because they have no effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval. The proposed changes to TS 5.5.14 are considered administrative in nature based on the revision to 10 CFR 50.59.

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

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

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

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of application for amendment: October 29, 1999, as supplemented June 21, and September 8, 2000

Brief description of amendment: The amendment revised the Technical

Specifications (TSs) to include: (1) the addition of operating limits for make-up tank (MUT) level and pressure; (2) the addition of surveillance requirements for the MUT pressure instrument channel; and (3) the revision of the calibration frequency for the MUT level instrument channel, the high- and low-pressure injection flow instrument channels, and the borated water storage tank instrument channel from ''Not to exceed 24 months'' to ''Refueling interval.'' Minor editorial changes and associated Bases changes were also made.

Date of issuance: December 26, 2000.

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

Amendment No.: 227.

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

*Dates of initial notices in **Federal Register:*** December 15, 1999 (64 FR 70090) and July 12, 2000 (65 FR 43042). The September 8, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notices.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 26, 1999, as supplemented on January 20, 2000.

Brief description of amendment: The proposed amendment would revise Technical Specifications (TSs) associated with the degraded voltage trip and the under-frequency reactor trip surveillance tests. For the degraded voltage trip, the proposed amendment would revise the TS to specify detailed operator actions to be taken if the minimum conditions could not be met rather than simply stating ''Cold Shutdown.'' The 6.9 kV under-frequency and reactor trip surveillance tests currently combine voltage and frequency testing under one item. The proposed TS amendment would separate the 6.9 kV voltage testing from the frequency testing and specify separate test requirements. In addition, the proposed TS amendment would require more frequent testing of the 480 volt emergency bus undervoltage reactor trip.

Date of issuance: December 28, 2000.

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

Amendment No.: 214.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 28, 2000 (65 FR 10565) The January 20, 2000, submittal contained supplemental information that did not change the original no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 27, 2000.

Brief description of amendment: This amendment changes the expiration date of the Operating License to 40 years from the date of issuance of the license rather than the date of the construction permit. Specifically, the amendment changes the expiration date of the Operating License from ''midnight on March 14, 2007'' to ''midnight on March 24, 2011.''

Date of issuance: December 14, 2000.

Effective date: As of the date of issuance.

Amendment No.: 192.

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

*Date of initial notice in **Federal Register:*** May 17, 2000 (65 FR 31352).

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

No significant hazards consideration comments received: No. Other comments are addressed in the Commission's related Safety Evaluation.

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

Date of application of amendments: September 12, 2000; supplemented October 4, October 26, November 6, and December 8, 2000.

Brief description of amendments: The amendments revise the Technical Specification requirements related to the reroll repair process used to repair steam generator tubes. They also institute new license conditions.

Date of Issuance: December 15, 2000.

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

Amendment Nos.: 318/318/318.

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

*Date of initial notice in **Federal Register:*** October 4, 2000 (65 FR 59222) The supplements dated October 4, October 26, November 10, and December 8, 2000, provided clarifying information that did not change the scope of the September 12, 2000, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No

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

Date of amendment request: November 23, 1999, as supplemented by letter dated October 12, 2000.

Brief description of amendment: The amendment revises Technical Specifications to incorporate the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the River Bend Station, Unit 1, Technical Specifications.

Date of issuance: December 20, 2000.

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

Amendment No.: 115.

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

*Date of initial notice in **Federal Register:*** March 8, 2000 (65 FR 12291).

The October 12, 2000, supplemental letter provided additional information to support staff review of the original application, and did not affect the initial finding of no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 23, 1999, as supplemented by letter dated October 19, 2000.

Brief description of amendment: The amendment incorporated the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the Arkansas Nuclear One, Unit No. 2, Technical Specifications.

Date of issuance: December 18, 2000.

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

Amendment No.: 228.

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

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12291).

The October 19, 2000, supplemental letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 23, 1999, as supplemented by letter dated October 12, 2000.

Brief description of amendment: The amendment incorporated the use of American Society of Testing and Materials D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the facility's TS.

Date of issuance: December 27, 2000.

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

Amendment No.: 170.

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

Date of initial notice in ***Federal Register:*** March 8, 2000, (65 FR 12291).

The October 12, 2000, supplement provided clarifying information that did not expand the scope of the original **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc.,Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: November 23, 1999.

Brief description of amendment: The amendment incorporated the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the Grand Gulf Nuclear Station Technical Specifications.

Date of issuance: December 18, 2000.

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

Amendment No: 144.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12291)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 2000.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: April 23, 2000.

Brief description of amendment: The amendment revised Technical Specification (TS) Surveillance Requirement 4.6.4.2.b.4 by deleting the word ''immediately,'' in order to remove a timing restriction for the hydrogen recombiner post-operation resistance testing. As a result, the amendment allows the recombiner units to cool after an operational test run, and provides a more-reliable measurement of the resistance-to-ground of the electrical insulation.

Date of Issuance: December 27, 2000.

Effective Date: December 27, 2000.

Amendment No.: 169.

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

Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34746)

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 19, 2000.

Brief description of amendment: Revised the Technical Specifications (TS) to extend the applicability of the current reactor coolant system pressure/temperature limits and allowed heatup and cooldown rates to 21.7 effective full power years of operation.

Date of Issuance: December 28, 2000.

Effective Date: December 28, 2000.

Amendment No.: 112.

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

Date of initial notice in ***Federal Register:*** August 23, 2000 (65 FR 51354). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 18, 2000.

Brief description of amendments: The amendments consist of changes to the ACTION Statement 18 to allow operation of the units with both channels of undervoltage protection bypassed for up to 8 hours to allow performance of the monthly surveillance without placing the units in a condition not permitted by the Technical Specifications (TSs). In addition, the amendments authorize an administrative change to Item 7.b. of TS Tables 3.3–2, 3.3–3, and 4.3–2 modifying ''Degraded Voltage '' to ''Undervoltage'' to make it consistent with the Updated Final Safety Analysis Report description.

Date of issuance: December 20, 2000.

Effective date: December 20, 2000.

Amendment Nos. 209 and 203.

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

Date of initial notice in ***Federal Register:*** September 20, 2000.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 25, 2000, as supplemented November 20, 2000.

Brief description of amendment: This amendment modifies Technical Specification (TS) 3.8.1.1, ''Electrical Power System—A.C. Sources—Operating,'' by extending the allowed outage time (AOT) for Action a.2 of TS 3.8.1.1 from 72 hours to 14 days, provided the Millstone Unit 3 (MP3) station blackout diesel generator is available to supply Millstone Unit 2 (MP2) power, otherwise the AOT is only allowed to be extended for 7 days. This one-time change is needed to support the replacement of the MP2 4160-volt electrical cross-tie line from Millstone Unit 1 (MP1) with a cross-tie from MP3. The modification is being made due to the decommissioning of MP1.

Date of issuance: December 21, 2000.

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

Amendment No.: 251.

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

*Date of individual notice in **Federal Register:*** November 1, 2000 (65 FR 65344).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendment: August 25, 2000.

Brief description of amendment: The amendments authorize changes to the Millstone Nuclear Power Station, Unit Nos. 2 and 3 (MP2 and MP3) Final Safety Analysis Report (FSAR). Millstone Unit No. 1 (MP1) is being decommissioned. To support this activity, several modifications are required to modify/eliminate MP1 systems that support the operation of structures, systems, and components that are shared or common to MP2 and MP3. One of the separation projects entails the replacement of the existing MP1 to MP2 4160-volt cross-tie with a new MP3 to MP2 4160-volt cross-tie. Northeast Nuclear Energy Company has evaluated this proposed new cross-tie utilizing the criteria of 10 CFR 50.59 and determined that the modification involved four unreviewed safety questions (USQs). One USQ pertains to MP2 and three USQs pertain to MP3.

Date of issuance: December 21, 2000.

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

Amendment Nos.: 252 and 190.

Facility Operating License Nos. DPR–65 and NPF–49: Amendments authorize changes to the FSAR.

Date of initial notice in Federal Register: October 20, 2000 (65 FR 65345).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–423 and 50–336, Millstone Nuclear Power Station, Unit Nos. 2 & 3, New London County, Connecticut

Date of application for amendment: June 26, 2000.

Brief description of amendment: The amendments revise technical specifications (TSs) 3/4.1.3.1, ''Reactivity Control Systems, Movable Control Assemblies, Full Length CEA Position'' and 3/4.1.3.1, ''Reactivity Control Systems, Movable Control Assemblies, Group Heights.'' Specifically, the changes revise the frequency for determining the operability of each rod not inserted fully in the core for Units 2 and 3 and the Deviation Circuit for Unit 2 from once every 31 days to once every 92 days.

Date of issuance: December 27, 2000.

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

Amendment Nos.: 253 and 191.

Facility Operating License Nos. DPR–65 and NPF–49: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46011)

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 4, 2000, as supplemented August 31, October 5, and November 16, 2000.

Brief Description of amendment: The amendment (1) adds new sections to the Technical Specifications (TSs) addressing missed surveillance test requirements and establishing a TS Bases control program, (2) revises TS Chapter 6 to allow use of generic personnel titles in lieu of plant-specific titles, (3) allows an alternative when the radiation protection manager does not meet the qualifications of Regulatory Guide 1.8, (4) relocates sections of TS Chapter 6 pertaining to onsite and offsite review and special inspections to the Operational Quality Assurance Plan, and (5) corrects typographical errors.

Date of issuance: December 21, 2000.

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

Amendment No.: 115.

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

*Date of initial notice in **Federal Register:*** May 31, 2000 (65 FR 34749).

The August 31, 2000, supplement provided updated TS pages to reflect incorporation of Amendment No. 110, which was issued subsequent to the May 4, 2000, application. In addition, a minor change in the proposed TS wording was proposed for consistency with the current TS. The October 5, 2000, supplement provided clarifying information to the May 4, 2000, application. The November 16, 2000, supplement proposed a minor wording change to be consistent with the latest revision of Standard TSs, NUREG–1433. The supplements were within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 2000.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: May 15, 2000

Brief description of amendments: The amendments revise station technical specification TS.3.7.B.6 to explicitly allow de-energizing motor control center (MCC) 1T1 or MCC 1T2 for up to 72 hours to accommodate installation of transfer switches for the MCCs.

Date of issuance: December 15, 2000.

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

Amendment Nos.: 155 and 146.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 12, 2000 (65 FR 43049)

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 29, 1999, as supplemented on November 21, 2000

Brief description of amendments: The amendments modify the Salem Unit Nos. 1 and 2 Technical Specifications (TS), and revise requirements stated in Notes 1 and 2 to Table 2.2–1, ''Reactor Trip System Instrumentation Setpoints,'' in order to add a tolerance associated with the setpoint values for the derivative module time constants (the Tau values) of the Over-Power, and the Over-Temperature delta temperature units.

Date of issuance: December 19, 2000.

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

Amendment Nos.: 239 and 220.

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

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4289).

The November 21, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 22, 2000 (PCN–520).

Brief description of amendments: The amendments revise Technical Specifications (TSs) 3.1.10, 3.3.9, 3.3.13, 3.4.5, 3.4.6, 3.4.7, 3.4.8, 3.8.2, 3.8.5, 3.8.8, 3.8.10, 3.9.2, 3.9.4 and 3.9.5 to allow small, controlled, safe insertions of positive reactivity while in shutdown modes.

Date of issuance: December 20, 2000.

Effective date: December 20, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—175; Unit 3—166.

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

Date of initial notice in ***Federal Register:*** October 13, 2000 (65 FR 60984).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 31, 2000 (TS 00–05).

Brief description of amendments: These amendments revised the Technical Specifications (TSs) by relocating various reactivity control system requirements from the TSs to the Sequoyah Technical Requirements Manual.

Date of issuance: December 18, 2000.

Effective date: December 18, 2000.

Amendment Nos.: 264 and 255.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** October 4, 2000 (65 FR 59226).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 2000.

No significant hazards consideration comments received: No.

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

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

Brief description of amendments: These amendments revised the Technical Specifications (TSs) by adding new requirements for maintaining soluble boron in the spent fuel pool.

Date of issuance: December 19, 2000.

Effective date: December 19, 2000.

Amendment Nos.: 265 and 256.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** October 18, 2000 (65 FR 62392).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 3rd day of January 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–596 Filed 1–9–01; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 24820; 812–11758]

Frank Russell Investment Company, et al; Notice of Application

January 3, 2001.

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

ACTION: Notice of an application for an order under sections 6(c) and 17(b) of the Investment Company Act of 1940 (''Act'') for an exemption from section 17(a) of the Act, under section 6(c) for an exemption from section 17(e) of the Act and rule 17e–1 under the Act, and under section 10(f) of the Act for an exemption from section 10(f).)

SUMMARY OF THE APPLICATION: Applicants request an order to permit certain registered open-end management investment companies advised by several investment advisers to engage in principal and brokerage transactions with a broker-dealer affiliated with one of the investment advisers and to purchase securities in offerings underwritten by a principal underwriter of which one of the investment advisers is an affiliated person. The transactions would be between a broker-dealer or principal underwriter and a portion of the investment company's portfolio not advised by the adviser affiliated with the broker-dealer or principal underwriter. Applicants also request relief to permit a portion of the portfolio to purchase securities in offerings underwritten by a principal underwriter of which the investment adviser to that portion is affiliated if the purchase is in accordance with all of the conditions to rule 10f–3 under the Act, except for the provision that would require aggregation of certain purchases.

APPLICANTS: Frank Russell Investment Company (''FRIC''), Russell Insurance Funds (''RIF''), and Frank Russell Investment Management Company (''Adviser'').

FILING DATES: the application was filed on August 24, 1999, and amended on December 1, 1999, and December 14, 2000.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be