individual or cumulative occupational radiation exposure, and (5) the probability of high trajectory turbine missiles impacting the spent fuel pool target area has been found to be so insignificant that the event need not be further considered as a design basis event.

On February 29, 1988 (53 FR 6041), the staff published ''Extended Burnup Fuel Use in Commercial LWR's; Environmental Assessment and Finding of No Significant Impact.'' This generic environmental assessment of extended fuel burnup in light water reactors found that ''no significant adverse effects will be generated by increasing the present batch-average burnup level of 33 GWD/MTU to 50 GWD/MTU or above as long as the maximum rod average burnup level of any fuel rod is no greater than 60 GWD/MTU.'' In addition, the environmental impacts of transportation resulting from the use of higher enrichment fuel and extended irradiation were published and discussed in the staff assessment entitled, ''NRC Assessment of the Environmental Effects of Transportation Resulting from Extended Fuel Enrichment and Irradiation,'' dated July 7, 1988. That assesement was published in connection with an Environmental Assessment related to the Shearon Harris Nuclear Plant, Unit 1, which was published in the **Federal Register** (53 FR 30355) on August 11, 1988, as corrected on August 24, 1988 (53 FR 32322). In these assessments, collectively, the staff concluded that the environmental impacts summarized in Table S–3 of 10 CFR 51.51 and in Table S–4 of 10 CFR 51.52 for a burnup level of 33 GWD/MTU are conservative and bound the corresponding impacts for burnup levels up to 60 GWD/MTU. These findings are applicable to the proposed action at Kewaunee which will limit burnup to 60 GWD/MTU.

With regard to potential non-environmental impacts, the proposed action involves components located entirely within the restricted area as defined by 10 CFR part 20. It does not affect non-radiological plant effluents and has no other environmental impact. The proposed action does not involve any of the historic sites located in the vicinity of Kewaunee as identified in Section II.C of the Kewaunee Final Environmental Statement. Therefore, there are no significant non-radiological environmental impacts associated with the proposed action.

Accordingly, the Commission concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission concluded that there are no significant environmental effects that would result from the proposed action, any other alternative would have greater environmental impacts and need not be evaluated.

The principal alternative would be to deny the requested amendment. This would not reduce the environmental impact of plant operations and would result in reduced operational flexibility.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement which was issued December 20, 1972.

Agencies and Persons Consulted

In accordance with its stated policy, on November 19, 1998, the staff consulted with Sarah Jenkins, an official of the Public Service Commission of the State of Wisconsin, 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 staff concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the staff has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's application dated April 15, 1998, as supplemented by letters dated July 27, August 13, September 28, and November 24, 1998, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW, Washington, D.C., and at the local public document room located at the University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311–7001.

Dated at Rockville, Maryland, this 25th day of November 1998.

For the Nuclear Regulatory Commission.

William O. Long, Sr.

Project Manager, Project Directorate III–1, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–32115 Filed 12–1–98; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 6, 1998, through November 19, 1998. The last biweekly notice was published on November 18, 1998 (63 FR 64106).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period.

However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By January 4, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: November 11, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to correct Surveillance Requirements (SRs) 3.6.11.6 and 3.6.11.7 and the associated Bases. These SRs currently are incorrect and do not reflect the Containment Pressure Control System (CPCS) as designed. Therefore, the proposed amendments would only revise the SRs; no change to the CPCS design is involved.

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:

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no significant effect on accident probabilities or consequences.

The CPCS is not an accident initiating system; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the CPCS is not being modified by this proposed amendment. The amendment merely aligns [TS] surveillance requirements with the existing design and function of the system. Therefore, there will be no impact on any accident 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. This amendment request does not impact any plant systems that are accident initiators, since the CPCS is an accident mitigating system.

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 proposed amendment. The CPCS is already capable of performing as designed. No safety margins will be impacted.

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

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

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

NRC Project Director: Herbert N. Berkow.

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 amendment request: October 15, 1998.

Description of amendment request: The proposed amendments would revise the pressure-temperature limits in the Technical Specifications for Units 1, 2, and 3. The proposed amendments would revise the heatup, cooldown, and inservice test limitations for the reactor coolant system of each unit to a maximum of 26 effective full-power years. The proposed amendments would also revise the Technical Specification for low temperature overpressure protection to reflect the revised pressure-temperature limits of the reactor vessels.

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

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

NO.

Each accident analysis addressed in the Oconee UFSAR [Updated Final Safety Analysis Report] has been examined with respect to the changes to the Reactor Pressure Vessel (RPV) pressure-temperature limit curves and related Low Temperature Overpressure settings. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of a DBA affected by this change. The revised pressure-temperature limits, which were developed based on NRC approved methodology or ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] Case N–514 as described in the Technical Justification, are not considered to be an initiator or contributor to any accident analysis addressed in the Oconee UFSAR. The added requirement to deactivate one pressurizer heater bank during low temperature operation does not significantly change the probability or consequence of any accident previously analyzed. No existing Technical Specification requirements are being deleted with this revision.

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

NO.

This license amendment revises Oconee RPV pressure-temperature limits. The revised pressure-temperature limits were developed based on NRC approved methodology or ASME Code Case N–514 as described in the Technical Justification. Operation of Oconee in accordance with these proposed new Technial Specifications will not create any failure modes not bounded by previously evaluated accidents. Consequently, this change will not create the possibility of a new or different accident from any accident previously evaluated.

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

NO.

This license amendment revises Oconee RPV pressure-temperature limits. The revised pressure-temperature limits were developed based on NRC approved methodology or ASME Code Case N–514 as described in the Technical Justification. The purpose of this license amendment is to assure that sufficient operating margin to safety is maintained in the operation of the Oconee reactor pressure vessels by establishing new, more limiting pressure-temperature limit curves and adding the requirement to deactivate one pressurizer heater bank. No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted. Therefore, there will be no significant reduction in any margin of safety.

Duke [Duke Energy Corporation] has concluded based on this information that there are no significant hazards considerations involved in this amendment request.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

NRC Project Director: Herbert N. Berkow. Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: November 11, 1998.

Description of amendment request: The proposed amendment would modify License Condition 2.C(9) to allow, on a one time only basis, an extension to the steam generator inspection interval of technical specification surveillance 4.4.5.3.b. This would allow the steam generator inspection interval to coincide with the 13th refueling outage or the end of 500 effective full power days, whichever occurs sooner.

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

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

The proposed change is temporary and allows a one time extension of specific surveillance requirements for Cycle 13 to allow surveillance testing to coincide with the 13th (1R13) refueling outage. The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of a system to perform its design function. Current monitoring of plant conditions and the surveillance monitoring required during normal plant operation will be performed as usual to assure conformance with technical specification operability requirements.

The technical specification steam generator tube inspection is intended to prevent the Steam Generator Tube Rupture analyzed in [Updated Final Safety Analysis Report] UFSAR Section 14.2.4 by maintenance of the integrity of the primary to secondary coolant boundary represented by steam generator tubes. The process by which this integrity is maintained is inspection of steam generator tubes at prescribed intervals, and the removal of defective tubes from service. Inspection intervals are based on preventing corrosion growth from exceeding tube structural limits, thereby preventing tube failure. The 1997 steam generator inspection characterized existing steam generator tube degradation, and degraded tubes were removed from service at that time. Degradation growth rates were evaluated for the next operating interval and it was determined that the steam generator tube structural integrity is maintained. Degradation of steam generator tubes was prevented during the extended outage by a carefully controlled, corrosion prevention program.

The proposed change does not affect the UFSAR and is consistent with changes granted for other plants. The surveillance extension does not involve a change to plant equipment and does not affect the performance of plant equipment used to mitigate an accident. This change, therefore, 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?

Extending the surveillance interval for the performance of specific inspections will not create the possibility of any new or different kind of accidents. No change is required to any system configurations, plant equipment or analyses.

Steam generator tube inspections determine tube integrity and provide reasonable assurance that a tube rupture or primary to secondary leak will not occur. Accidents involving steam generator tube rupture are analyzed in UFSAR Section 14.2.4, ''Steam Generator Tube Rupture.'' The only type of accident that can be postulated from extending the steam generator inspection interval would be a tube leak or rupture which are analyzed in the UFSAR. No new failure modes are created by the surveillance extension. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Surveillance interval extensions will not impact any plant safety analyses since the assumptions used will remain unchanged. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance interval is being extended. Based on engineering judgement, extending the surveillance interval for the performance of these specific inspections does not involve a significant reduction in the margin of safety derived from the required surveillances.

The margin of safety depends upon maintenance of specific operating parameters within design limits. In the case of steam generators, that margin is maintained through assurance of tube integrity as the primary to secondary boundary. Assurance of tube integrity is provided through periodic in-service inspection of tubes and removal of defective tubes from service. Additional margin is provided through protection from possible consequences of steam generator tube failure by mitigation systems. Radiation monitors provide a detection capability of primary to secondary leakage to enable a prompt response. Maintenance of the steam generator water chemistry in accordance with [Electric Power Research Institute] EPRI guidelines provides additional margin of safety. Therefore, the plant will be maintained within the analyzed limits and the proposed extension will not significantly reduce the margin of safety.

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

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: June 30, 1998.

Description of amendment request: The proposed change modifies the Engineered Safety Features Actuation System (ESFAS) portion of the Arkansas Nuclear One, Unit-2 (ANO–2) Plant Protection System (PPS). This modification is designed to defeat the backup power supply for the auctioneered power sources for channel A and D Reactor Protective System (RPS) and ESFAS bistables, and to provide selective logic for Emergency Feedwater Actuation Signals and Main Steam Isolation Signals. This will ensure that ESFAS will have the redundancy and independence sufficient to assure that (1) no single failure results in loss of the protection function with a channel in indefinite bypass, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy required by the ANO–2 Technical Specifications (TSs). The proposed modification to the ANO–2 PPS has been determined to involve an Unreviewed Safety Question in accordance with 10 CFR 50.59(a)(2).

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

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

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

The ANO–2 Plant Protection System (PPS) includes the electrical and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating signals associated with the two protective functions, Engineered Safety Feature Actuation System (ESFAS) and Reactor Protective System (RPS). The RPS is that portion of the PPS which generates signals that actuate a reactor trip. The ESFAS is that portion of the PPS which generates signals that actuate Engineered Safety Features (ESF) to mitigate the consequences of an accident.

The ANO–2 Safety Analysis Report (SAR) section 15.1.31 ''Loss Of One DC System'' analyzes failure of a DC bus (FODCB) as initiator and its causes. The causes for the FODCB are DC leg to leg fault in the bus or in the power distribution circuit from the battery. Since the proposed change has no impact on the accident initiator, the frequency of occurrence is not changed. In order for the FODCB as a single failure with an accident to de-energize two [Vital Instrument Buses (]VIBs[)], the FODCB would have to occur prior to the safety bus energization by offsite bus fast transfer or prior to safety bus energization by the emergency diesel generator (EDG). The potential for de-energization of one pair of VIBs is, therefore, limited to time from initiation of the accident to time for safety bus response to the secondary plant and Reactor Protective System trips.

The effects of the FODCB are being revised to assume a secondary plant trip that results in de-energization of one power division. The existing analysis conclusions remain unchanged. The accident analysis is being revised to include de-energization of a pair of vital AC instrument channels. De-energization of two vital AC sources has not been previously documented as a design bases event.

Auctioneered bistable power supplies for Plant Protection System (PPS) channels A and D are being modified to a single power source for each of these two channels. Single channel trips will result for all PPS functions in channels A or D for loss of its single channel bistable power source. The PPS channels B and C auctioneered power supplies remain unchanged to maintain Recirculation Actuation Signal (RAS) response to a FODCB.

Regarding PPS measurement channels with increasing signal setpoints, de-energization of a single power supply either results in failure of a measurement channel (B or C) to a non-tripped state or in failure of a measurement channel (A or D) to a tripped state. Neither single channel failure scenario impacts accident initiation or mitigation. For PPS measurement channels with decreasing signal setpoints the single channel de-energization events result in failure of a single affected measurement channel to a tripped state. The PPS two out of three logic design with a channel bypassed ensures operability with a single channel failure. Neither condition impacts accident frequency or consequences.

With the exception of Recirculation Actuation Signal (RAS) and Emergency Feedwater Actuation Signal (EFAS), a FODCB results in an automatic ESFAS initiation for those functions with decreasing signal setpoints. For other ESFAS functions with a decreasing signal, channels A and C or channels B and D fail to the tripped state. For those functions with an increasing signal setpoint (including EFAS), a FODCB results in a single channel failing not tripped, one channel tripping, and two channels remaining functional. System level functions remain operable with either a one out of two logic (no channels bypassed) or a one out of one logic (with a channel bypassed).

Interposing relay actuation logic has changed from single trip path to selective trip path logic. This change insures emergency feedwater (EFW) discharge valves will receive an automatic open or close demand based on steam generator level and pressure demands.

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

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

In response to de-energization of a pair of Vital Instrument Buses (VIBs), those ESFAS functions with increasing signal setpoints, as a minimum, remain functional with one out of one logic. One channel trips, one channel does not trip, and two channels remain functional. One of the functional channels may be bypassed without impact on operability. The trip response of those ESFAS functions with decreasing signal to trip setpoints remains unchanged.

EFAS coincidence logic to close the EFW discharge valves requires three out of four channels to be in a non-tripped state. With a FODCB one channel is tripped, one channel is not tripped, and two channels are functional. The close logic becomes two out of two with a FODCB.

By defeating the auctioneered bistable power sources for PPS channel A and D bistables, PPS measurement channel A or D will fail to its tripped state. This change ensures no more than one channel (B or C) fails to a non-tripped state for the FODCB.

With selective logic EFAS pump discharge valves will receive control signals to initiate emergency feedwater and to terminate emergency feedwater flow by open and close demands generated independent of the 120 Volt channel pair de-energization.

The existing ANO–2 Failure Modes and Effects Analysis does not document failure of a pair of vital instrument AC channels. Neither the 120 Volts AC nor the 125 Volt DC system single failure analysis assumes failure of two channels of 120 Volts AC. Even though the failure of either pair of VIBs caused by a FODCB is not a result of the proposed change, the SAR change will address the potential for de-energization of a pair of instrument buses. The ANO–2 SAR will be updated to reflect the documentation and modification of the PPS design to ensure safe plant response.

Even though the plant response to FODCB is being modified, the proposed ANO–2 PPS design resolution does not create the possibility of a new or different kind of accident from any previously evaluated in the SAR. The PPS will have the redundancy and independence sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy required by the TS. PPS will also meet the single failure criterion of IEEE 279–1971 to the extent that any single failure within the system does not prevent proper protective action at the system level and no single failure will defeat more than one of the four protective channels associated with any one trip function.

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

Technical Specification Bases 3/4.3.1 & 3/4.3.2 assure sufficient PPS redundancy is maintained to permit a channel to be bypassed. Under the current design, a FODCB will result in reduction of margin by decreasing the number of functional channels to less than two. However, with the proposed modification removal from service of any component or channel for indefinite bypass will not result in loss of the minimum redundancy required by the TS. This activity will restore the margin by ensuring ESFAS required functions remain capable of automatic actuation with a FODCB.

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, Entergy Operations has determined that even though the proposed PPS design description results in an accident or malfunction of a different type, the requested change does *not* involve a significant hazards consideration.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: October 29, 1998.

Description of amendment request: The proposed amendment would revise the terminology used in the St. Lucie Plant Technical Specifications (TS) relative to the implementation and automatic removal of certain reactor protection system trip bypasses to ensure that the meaning of explicit terms used in the TS are consistent with the intent of the stated 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) 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 proposed amendments are administrative in nature, and do not change the function or the setpoints of the RPS trip bypass features. The revisions simply make corrections to the Notation of TS Tables 2.2–1 and 3.3–1 to ensure that the meaning of explicit terms used in the Notes is consistent with the intent of the stated requirements based on the St. Lucie plant design. The proposed technical specification changes do not involve accident initiators, do not change the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, and do not alter any conditions assumed in the plant accident analyses. Therefore, operation of either facility in accordance with its 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 accident previously evaluated.

The proposed amendments are administrative in nature and will not change the physical plant or the modes of plant operation defined in the facility operating licenses. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments are administrative in nature and do not change the function or the setpoints of the RPS trip bypass features. The revisions simply make corrections to the Notation of TS Tables 2.2–1 and 3.3–1 to ensure that the meaning of explicit terms used in the Notes is consistent with the intent of the stated requirements based on the St. Lucie plant design. The proposed changes do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration. This notice is intended to replace an exigent notice of consideration of issuance of amendment for St. Lucie Unit 1, previously published as exigent TS amendments for both St. Lucie Units 1 and 2 in the **Federal Register** (63 FR 59809). The amendment request for St. Lucie Unit 2 will continue to be considered as an exigent amendment as noticed in the **Federal Register** (63 FR 59809).

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: November 10, 1998.

Description of amendment request: The proposed Technical Specification (TS) change would remove the restriction on the sale or lease of property within the exclusion area and replace the restriction with a requirement to retain complete authority to determine and maintain sufficient control of all activities including the authority to exclude or remove personnel and property within the minimum exclusion distance. A TS Bases page for the proposed change is included. Also included are clarifications and administrative changes which (1) clarify TS definition 1.38 to become ''Site Boundry'' from the current term ''Exclusion Area'' to be consistent with 10 CFR 20.1003 definition for Site Boundry and the 10 CFR 100.3 definition of Exclusion Area, (2) convert the one occurrence of the use of TS definition from Exclusion Area to Site Boundry in TS 6.8.4(a)(9), and (3) revise and update the Table of Contents for Section I Definitions.'

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

The proposed change is administrative in nature and does not affect the purpose, function, performance, operability or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

[Technical Specification Change Request] TSCR 264 does not change the size or location of the exclusion area. Since the exclusion area size and location are not being changed and no physical or procedural changes are being made to the plant, radiological consequences in the exclusion area are not affected by this TSCR.

This change addresses the existing technical specification restriction on the sale or lease of property within the ''exclusion area'' by ensuring that the licensee will retain at all times the complete authority to determine and maintain sufficient control of all activities through ownership, easement, contract and/or other legal instruments on property within the minimum exclusion distance including the authority to exclude or remove personnel and property within the minimum exclusion distance.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the proposed change requires the licensee to retain complete authority and sufficient control of all activities in the exclusion area, 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. Would operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The p[ro]posed change is administrative in nature and does not affect the purpose, function, performance, operability or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

This change addresses the existing technical specification restriction on the sale or lease of property within the ''exclusion area'' by ensuring that the licensee will retain at all times the complete authority to determine and maintain sufficient control of all activities through ownership, easement, contract and/or other legal instruments on property within the minimum exclusion distance including the authority to exclude or remove personnel and property within the minimum exclusion distance.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the proposed change requires the licensee to retain complete authority and sufficient control of all activities in the exclusion area, 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 accident previously evaluated.

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

The p[ro]posed change is administrative in nature and does not affect the purpose, function, performance, operability or testing of and does not make any physical or procedural changes to plant systems, structures or components. Also, all existing technical specification limiting conditions for operation and surveillance requirements are retained.

This change addresses the existing technical specification restriction on the sale or lease of property within the ''exclusion area'' by ensuring that the licensee will retain at all times the complete authority to determine and maintain sufficient control of all activities through ownership, easement, contract and/or other legal instruments on property within the minimum exclusion distance including the authority to exclude or remove personnel and property within the minimum exclusion distance.

Therefore, since no physical or procedural changes are being made to existing plant systems, structures or components and since the proposed change requires the licensee to retain complete authority and sufficient control of all activities in the exclusion area, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in margin of safety.

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

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

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

NRC Project Director: Cecil O. Thomas.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2 (NMP2), Oswego County, New York

Date of amendment request: October 16, 1998.

Description of amendment request: The proposed amendment would make the following revisions to Technical Specifications (TSs) 3/4.7.1.1: (1) Ensure that four service water (SW) pumps are operating with the divisional cross connect valves open during Operational Condition 1, 2 and 3 (current TS requires two SW pumps associated with one loop to be operating); (2) Increase the number of division 1 and 2 heaters required to be operable from 7 per division per intake to 14 per division per intake; (3) The actions necessary for having less than the required equipment is being revised to reflect the new limits for SW equipment; and (4) SW supply header discharge water temperature is being increased from 81 to 82 °F. TS 3.7.1.2, Table 3.3.9–1, and Table 4.3.9.1–1 are revised to add ''when handling irradiated fuel in the secondary containment'' to the applicability section. Table 3.3.9–1 is being revised to decrease the temperature at which the Intake Deicing Heaters are required to be in service from 39 to 38 degrees F. TS 3.7.1.2 proposed change is to specify that the necessary portions of the SW system needed to support equipment required to be operable shall be operable; the Action Section proposed revision reflects this change. TS 4.7.1.2.1 surveillance requirement proposed change is to increase the flow rate of SW pumps from 6500 GPM to 9000 GPM and to change the SW pumps pressure from 80 psi discharge pressure to 70 psi differential pressure; TS 4.7.1.2.2 is being revised to decrease the intake tunnel water temperature from 39 to 38 degrees F. The surveillance for the Intake Deicing Heaters is being changed to reflect the increase in the number of heaters required. The title of ''Plant Service Water System'' is being changed to ''Service Water System.''

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

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

The SW System is a once-through system which supplies water from Lake Ontario to various essential and non-essential components, as required, during normal plant operation and shutdown conditions. The System is designed with suitable redundancy to provide a reliable source of cooling water for the removal of heat from essential plant components, including the RHR [residual heat removal] heat exchangers, the EDGs [emergency diesel generators], and room coolers for ECCS [emergency core cooling system] equipment, which are required for safe reactor shutdown following a LOCA.

LCO 3.7.1.1 and LCO 3.7.1.2 each currently requires two independent SW System loops to be operable, with one of the loops in operation. The current LCOs do not provide adequate guidance regarding the minimum number of operating pumps. NMPC [Niagara Mohawk Power Corporation] proposes to revise LCO 3.7.1.1 and its associated Actions and SRs to provide assurance that four SW pumps are operable and are operating within acceptable system parameters, with the divisional cross-connect valves open, during Operational Conditions 1, 2, and 3 to meet the limiting LOCA analysis assumptions.

TS Section 3/4.7.1 currently specifies a maximum SW supply header discharge water temperature of 81 degrees F and a limiting temperature for Intake Deicing Heater ystem operability (intake water) temperature of 39 degrees F. In addition, TS Table 3.3.9–1, Action 144, requires the Intake Deicing Heater System heaters to be placed in service when the Lake Ontario water temperature reaches 39 degrees F. NMPC proposes to revise Action 144 of TS Table 3.3.9–1 and TS LCO 3.7.1.1, including its associated Actions and SRs [surveillance requirements], to increase the supply header discharge water temperature to its analytical limit of 82 degrees F and reduce the limiting temperature for the Intake Deicing Heater System Action and operability requirements to 38 degrees F.

Appropriate changes to LCO 3.7.1.2 and its associated Actions and SRs are also proposed in order to assure consistency with the SW System analyses assumptions during shutdown conditions. The current LCO Actions do not account for the varying flows and heat loads that may be required for various plant shutdown conditions. The revision to the Applicability for LCO 3.7.1.2 and TS Tables 3.3.9–1 and 4.3.9.1–1 will assure that the SW System is operable during periods when irradiated fuel is being handled in the secondary containment and essential loads cooled by the SW System are required to be operable (e.g., EDG). A footnote has been added to define Operational Condition * and is consistent with similar footnotes in the TSs. The proposed changes will assure that the necessary ortions of the SW System and the necessary Divisions of the Intake Deicing Heater System heaters are operable that are supporting equipment required to be operable.

It is further proposed to change the system title identified in the Index and in TS Section 3/4.7.1, including the LCOs and SRs, from ''Plant Service Water System'' to ''Service Water System'' to be consistent with the NMP2 [Nine Mile Unit 2] UFSAR [Updated Final Safety Analysis Report].

The changes do not involve any physical alteration of the plant, and the SW System will remain capable of providing sufficient cooling flow for the essential cooling loads during plant operation and also during plant shutdown. The changes will have no impact on the design or function of the SW System and its components, thus assuring that the characteristics and functional performance are maintained consistent with the event precursors and the conditions and assumptions of the current design basis accident and transient analyses. The changes to the LCO AOTs [allowed outage times] are either consistent with or are more conservative than the current AOTs. Based on the above, adequate assurance is provided that the probability of event initiation will remain as previously analyzed. Maintaining four pumps operating within acceptable system parameters, with the divisional cross connect valves open, during Operational Conditions 1, 2, and 3 provides assurance that the essential functions supported by the SW System are maintained. Particularly, adequate SW flow assures that the primary and secondary containments can perform their intended functions of limiting the release of radioactive materials to the environment following a LOCA. The small (1 degree F) change in the SW supply header discharge water (UHS) temperature and Intake Deicing Heater System actuation temperature maintain the current design basis for the UHS and SW Systems such that there will be no impact on the LOCA analyses assumptions or conclusions. The proposed changes to the SW System TSs do not adversely affect the capability of plant systems, structures, and components to respond to any accident in Operational Conditions 4, 5, and *. As a result, there will be no degradation of the primary or secondary containment or any other fission product barriers which could increase the radiological consequences of an accident. In addition, other essential accident mitigation equipment supported by the SW System will not be adversely impacted. It is, therefore, concluded that operation of NMP2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes do not result in any hardware changes or physical alteration of the plant which could introduce new equipment failure modes, and there will be no impact on the design or function of the SW System or its components. The primary and secondary containment post-LOCA responses remain within previously assessed limits of temperature and pressure. Furthermore, adequate cooling flow is assured during plant operation and also during shutdown conditions such that essential systems and components remain within their applicable design limits. It is, therefore, concluded that no requirements are eliminated or new requirements imposed which could affect equipment or plant operation such that new credible accidents are introduced. Accordingly, operation of NMP2, 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 operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The changes provide assurance that the SW System will remain capable of providing sufficient cooling flow for the essential cooling loads during plant operation and also during plant shutdown such that essential systems and components remain within their applicable design limits. The changes will have no impact on the design or function of the SW System and its components, thus assuring that the characteristics and functional performance are maintained consistent with the conditions and assumptions of the current design basis accident and transient analyses. Maintaining four pumps operating within acceptable system parameters, with the divisional cross connect valves open, during Operational Conditions 1, 2, and 3 provides assurance that post-LOCA radioactive releases are maintained within 10 CFR 100 limits. The small (1 degree F) change in the SW supply header discharge water (UHS) temperature and the limiting temperature for the Intake Deicing Heater System Action and operability requirements maintains the current design basis for the UHS and SW Systems such that there will be no impact on the LOCA analyses assumptions or conclusions.

These changes will not result in a reduction in margin to the System analytical limits. Furthermore, maintaining the intake bar surface temperature at least 1 degree F above freezing provides an adequate margin to prevent the adherence of ice, and provides assurance that sufficient flow area is always heated such that the SW System will remain capable of providing adequate cooling flow in the event of a LOCA. Similarly, maintaining the required SW System flow and temperature during Operational Conditions 4, 5, and * will assure that the associated equipment is operable such that radioactive releases are maintained within 10 CFR 100 limits. It is, therefore, concluded that the changes do not eliminate any requirements, impose any new requirements, or alter any physical parameters which significantly reduce the margin to an acceptance limit or adversely affect the margins associated with the fission product barriers as established by the design basis accident and transient analyses. Accordingly, operation of NMP2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa.

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

Date of amendment request: September 28, 1998.

Description of amendment request: The proposed amendment would change Technical Specifications 3.3.2.1, ''Instrumentation—Engineered Safety Features Actuation System''; 3.4.6.2, ''Reactor Coolant System—Reactor Coolant System Leakage''; 3.4.8, ''Reactor Coolant System—Specific Activity''; 3.6.2.1, ''Containment Systems—Depressurization and Cooling Systems Containment Spray and Cooling Systems''; 3.6.5.1, ''Containment Systems—Secondary Containment Enclosure Building Filtration System''; 3.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System''; and 3.9.15, ''Refueling Operations—Storage Pool Area Ventilation System—Fuel Storage.'' Information would also be added to the Bases of the associated Technical Specifications to address the proposed changes.

The proposed amendment would also revise the Operating License DPR–65 by incorporating a change to the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). The change to the FSAR is associated with the revised main steamline break analyses, new determination of the radiological consequences of a main steamline break, and a revised determination of the radiological consequences of the design basis loss-of-coolant accidents (LOCAs).

The proposed changes to the main steamline break analysis, as described in the FSAR, are based on the revised Siemens Power Corporation steamline break methodology. The report describing the revised methodology was submitted by Siemens Power Corporation to the NRC for approval in a letter dated June 30, 1998. The revised methodology was used to perform the Millstone Unit No. 2 plant-specific analysis for post-scram main steamline break. This plant-specific analysis was submitted by NNECO in a letter dated August 12, 1998, which proposed to change the list of documents in the Technical Specifications that describe the analytical methods used to determine the core operating limits. The proposed changes contained in this letter assume approval of the previously submitted revised Siemens Power Corporation steamline break methodology, and the changes to the list of documents in the Millstone Unit No. 2 Technical Specifications that describe the analytical methods used to determine the core operating limits.

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:

In accordance with 10CFR50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

Analyses Changes

The main steam line break analyses and the determinations of the radiological consequences of the main steam line break and loss of coolant accident have been revised. A brief summary of the significant changes to the main steam line break analyses and the radiological consequences of the main steam line break and loss of coolant accident is presented below.

1. The limited fuel failure following a main steam line break outside containment results in an increase in the calculated radiological consequences both off-site and in the control room. To limit the consequences of a main steam line break outside containment, the

Technical Specification allowed steam generator tube leakage will be reduced to 0.035 gpm [gallons per minute] per steam generator.

2. Credit will now be taken for iodine removal from the containment atmosphere by the Containment Spray System (CSS). The use of the CSS for iodine removal has not been previously approved by the NRC.

3. The proposed increase to the allowable control room in-leakage will provide additional operational flexibility to address expected minor system degradation over time. The increase in the allowable control room in-leakage will result in an increase in the calculated dose to the Control Room Operators.

4. The addition of the dose consequences from containment sump backleakage to the Refueling Water Storage Tank (RWST) has been included in the off-site and control room loss of coolant accident (LOCA) analyses increases the consequences of previously evaluated accidents.

The containment sump backleakage into the RWST results in sump water entering the RWST when the RWST is at its minimum level. The RWST will become a radioactive source and contribute a shine dose to the surrounding areas. The increase in dose rates onsite will not prevent operators from remaining in the control room or from accessing equipment needed to mitigate the accident.

All piping and valves associated with RWST backleakage are located in a harsh radiation area. Backflow from the sump might increase dose rates in the area where these components are located. Additional dose contributions, where they occur, do not adversely impact the environmental qualification of the vital equipment located there. All vital equipment would continue to perform its safety function.

5. Credit will be taken in the main steam line break analyses for the recently installed cavitating venturis in the Auxiliary Feedwater System. However, this will not change the amount of fuel failure. Therefore, credit for this equipment will not impact the radiological consequences of a main steam line break.

6. Credit will be taken for the Reactor Coolant System (RCS) low flow reactor trip for the pre-scram inside containment main steam line break analysis. This equipment will be qualified for the expected containment environment following a main steam line break inside containment and will be added to the Environmental Qualification Master List.

7. Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1. [Footnote—B.D. Kenyon letter to the NRC, ''Millstone Nuclear Power Station, Unit No. 1 Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor,'' dated July 21, 1998.]

The revised main steam line break analyses and the revised determinations of the radiological consequences of the main steam line break and design basis LOCA analyses take credit for equipment not previously assumed in the analyses, and for plant or equipment operating restrictions not currently contained in the Technical Specifications. The changes to the analyses will not adversely affect the probability of an accident previously evaluated, but the revised analyses results do indicate that the consequences of an accident previously evaluated will increase. Specifically, the following changes cause an increase in the consequences of an accident previously evaluated.

1. The increase in allowable control room in-leakage from 100 SCFM [standard cubic feet per minute] to 130 SCFM when the Control Room Emergency Ventilation System is operating in the recirculation/filtration mode.

The dose to the Control Room Operators from a Millstone Unit No. 2 LOCA increased from 9.25 to 25.8 rem to the thyroid and from 0.205 to 2.29 rem to the skin. The dose to the whole body decreased. (Both low wind speed and high wind speed release conditions were analyzed. The low wind speed condition bounds the high wind speed condition.) The dose to the Control Room Operators from a Millstone Unit No. 3 LOCA increased from 2.67 to 14 rem to the skin and from 0.209 to 1.484 rem to the whole body. The dose to the thyroid decreased. The doses to the Control Room Operators from either a Millstone Unit No. 2 or Unit No. 3 LOCA remain below the GDC [General Design Criterion] 19 criteria of 30 rem thyroid, 5 rem whole body and 30 rem to the skin.

The new calculated doses to the Millstone Unit No. 2 Control Room Operators from a main steam line break outside containment are 29 rem thyroid, 0.03 rem whole body and 0.5 rem skin. The doses to the Millstone Unit No. 2 Control Room Operators are below the GDC 19 criteria of 30 rem thyroid, 5 rem whole body, and 30 rem to the skin. (Note: The dose to the Control Room Operators from a main steam line break was not previously evaluated because fuel failure was not predicted to occur.)

2. The limited fuel failure that is predicted in the revised main steam line break analyses.

Previously, the radiological consequences of a main steam line break were not determined and were not presented in the FSAR because fuel failure was not predicted to occur. Because of the predicted limited fuel failure for the main steam line break outside of containment, the radiological consequences were analyzed. The results to the Exclusion Area Boundary (EAB) are 4.8 rem thyroid and 0.06 rem whole body. The results to the Low Population Zone (LPZ) are 2.3 rem thyroid and 0.02 rem whole body. To meet the dose acceptance criteria to the Millstone Unit No. 2 Control Room Operators, the maximum allowable Technical Specification primary to secondary leak rate is being reduced to 0.035 gpm per steam generator. The results to the Millstone Unit No. 2 Control Room Operators are 29 rem thyroid, 0.03 rem whole body and 0.5 rem skin. The main steam line break outside containment is the limiting accident for the Millstone Unit No. 2 Control Room Operators. However, the dose consequences of a main steam line break are less than the 10CFR100 limits off-site of 300 rem thyroid and 25 rem whole body, and the doses to the Millstone Unit No. 2 Control Room Operators are below the GDC 19 criteria of 30 rem thyroid, 5 rem whole body, and 30 rem to the skin.

3. Taking credit for the low RCS flow reactor trip for the pre-scram inside containment main steam line break analysis.

Previous analyses did not credit the low RCS flow reactor trip in a harsh environment. This credits the low flow trip in a manner not previously reviewed by the NRC for Millstone Unit No. 2. Without credit for this reactor trip, the predicted fuel failure for steam line breaks inside containment would be higher.

4. Taking credit for the removal of radioactive iodine from the containment atmosphere by containment spray.

Previous analyses did not rely on the spray function to reduce iodine concentration in the post-accident atmosphere inside containment. This adds a mitigation function to the CSS that has not been previously reviewed by the NRC for Millstone Unit No. 2. Without credit for the removal of iodine, the predicted dose consequences following a LOCA would be higher.

5. The addition of sump backleakage to the RWST during a LOCA.

The resultant dose contribution to the LPZ from RWST backleakage is 1.487 rem thyroid and 0.11 rem whole body. The total dose to the LPZ from a design basis LOCA is 21.86 rem thyroid and 0.941 rem whole body. The dose is well below the 10CFR100 limits of 300 rem thyroid and 25 rem whole body. The dose to the EAB was not affected because leakage into the RWST does not start until 25.45 hours post-LOCA and the EAB is a 2-hour dose.

The resultant dose contribution to the Millstone Unit No. 2 Control Room Operators from RWST backleakage is 3.75 rem thyroid, 0.017 rem whole body and 0.296 to the skin. The total dose to the Millstone Unit No. 2 Control Room Operators from the LOCA is 25.8 rem thyroid, 0.718 rem whole body and 2.29 rem to the skin. These doses are below the GDC 19 limits of 30 rem thyroid and skin, and 5 rem whole body.

The analyses results meet the guidance contained in SRP [Standard Review Plan] 15.1.5, SRP 15.6.5, and the limits of 10CFR100 and GDC 19. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification Changes

Technical Specification Non-Technical Changes

The minor editorial and non-technical changes to correct spelling (Technical Specification 3.3.2.1), modify the title of a table column (Technical Specification 3.4.8), clarify the type of measurement performed (Technical Specification 3.4.8), and establish consistent terminology (Technical Specification 3.7.6.1) will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.6.2

The reduction in the maximum allowable value of primary to secondary leakage per steam generator is consistent with the new radiological assessment of the potential control room operator exposure following a main steam line break outside of containment. The wording change to SR [Surveillance Requirement] 4.4.6.2.1 will clarify that the water inventory balance is used to verify compliance with the identified and unidentified leakage limits. Pressure boundary leakage would first show up as unidentified leakage during performance of SR 4.4.6.2.1. Further investigation, (plant walkdown) would be necessary to classify the unidentified leakage as pressure boundary leakage. This is consistent with established plant practices to detect pressure boundary leakage.

The addition of the new SR 4.4.6.2.2 will address the primary to secondary leakage limit. The new SR will include an exception to Technical Specification 4.0.4 that will allow the determination of primary to secondary leakage to be deferred until after Mode 4 is entered. Even though verification of compliance with the primary to secondary limit will not be done prior to entering Mode 4, the limit is still expected to be met.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.4.8

The addition of the words ''of gross specific activity'' to the Limiting Condition for Operation (LCO), Action Statements, and SR will clarify what the E-Bar limit applies to. This is consistent with the Technical Specification Definition (1.20) for E-Bar.

The addition of a footnote (*) to state the power history requirements for the determination of E-Bar will ensure that the necessary plant conditions are established prior to performing the analysis. This will not affect the E-Bar LCO limit or the requirement to perform the analysis. The proposed change is consistent with NUREG—0212 and NUREG—1432.

The footnote will also specify that the provisions of Specification 4.0.4 are not applicable. This will allow entry into Mode 1, without determining the value of E-Bar, assuming that the power history requirements will not be met until after Mode 1 is entered. This will normally only apply following an extended shutdown.

The Isotopic Analysis for Iodine (including I–131, I–133, and I–135) sample requirement will be expanded to include the LCO requirement for 100/E-Bar. This is consistent with the requirements of Action Statement d. This change will expand the sampling requirement for iodine. Minor wording changes will also be made to be consistent with the proposed changes to the LCO wording.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.6.2.1

The revised radiological assessment calculation for the design basis accident credits iodine removal from the containment atmosphere by the CSS. This will require a reduction in the allowed outage time (AOT) of one containment spray train from seven days to seventy two hours. This AOT is consistent with NUREG–0212 and NUREG–1432. This will help ensure that plant equipment assumed in the safety analyses will be available. This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.6.5.1

The value for the pressure drop across the combined HEPA [high-efficiency particulate air] filters and charcoal adsorber banks specified in SR 4.6.5.1.d.1 will be changed from a generic value [less than or equal to] 6 inches water gauge) to a plant specific value [less than or equal to] 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.7.6.1

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.7.6.1.e.1 will be changed from a generic value [less than or equal to] 6 inches water gauge) to a plant specific value [less than or equal to] 3.4 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation.

SR 4.7.6.1.e.2 will be expanded to clarify that the test of the capability of the Control Room Emergency Ventilation Trains to switch to the recirculation mode is performed with the trains initially operating in the normal mode and the smoke purge mode of operation. This will not affect the requirement that the trains be capable of switching to the recirculation mode.

The value of allowable control room air in-leakage specified in SR 4.7.6.1.e.3 will be increased from 100 SCFM to 130 SCFM. This is consistent with the recently revised control room radiological analysis for the design basis accidents.

The proposed increase will provide additional operational flexibility to address expected minor system degradation over time. This increase is supported by the new analysis.

The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.9.15

The value for the pressure drop across the combined HEPA filters and charcoal adsorber banks specified in SR 4.9.15.d.1 will be changed from a generic value [less than or equal to] 6 inches water gauge) to a plant specific value [less than or equal to] 2.6 inches water gauge). This is a more restrictive change which will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on how any of the associated systems or components function to prevent or mitigate the consequences of design basis accidents. Also, the proposed changes have no adverse effect on any design basis accident previously evaluated since the changes are consistent with the revised analyses, and the appropriate acceptance criteria are met for the revised analyses. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes.

Also, the response of the plant and the operators following these accidents is unaffected by the change. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Analyses Changes

The acceptance criteria for a main steam line break in the SRP 15.1.5 does not exclude the prediction of fuel failure. Instead, the SRP requires that ''Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling.'' The limited fuel failure that is now predicted in the revised main steam line break analyses meets this acceptance criterion. In addition, the RCS low flow reactor trip that is now being credited to function in a harsh environment to limit fuel failure is already required to be operable by Technical Specifications.

The revised dose consequences for the design basis accidents assumes a control room in-leakage of 130 SCFM. In addition, iodine removal by the CSS, which is already required to be operable by Technical Specifications, is assumed. The acceptance criteria for the dose consequences of the design basis accidents to the EAB, LPZ and the control room personnel is met in the revised analyses. Therefore, the revisions to the dose consequence analyses for the design basis accidents do not involve a significant reduction in the margin of safety.

Technical Specification Changes

The proposed changes will correct spelling and terminology errors, reduce the maximum allowable primary to secondary leakage, add a new surveillance requirement, modify surveillance requirements for RCS specific activity, reduce the allowed outage time for a containment spray train, reduce the allowed pressure drop across the control room and enclosure building HEPA [high-efficiency particulate air] filters, and increase the control room maximum allowed in-leakage. These changes will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

The only adverse impact of the proposed changes is that the dose consequences following an accident may increase. However, the revised analyses show that the acceptance criteria for the accident analyses are met. Therefore, based on the responses above, the proposed changes are deemed safe.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC. The minor editorial and non-technical changes proposed herein to correct reference, spelling, and terminology errors are enveloped by example (i), a purely administrative change to Technical Specifications. The changes proposed herein to add a new surveillance requirement to verify primary to secondary leakage and to reduce the allowable pressure drop across various ventilation filters are enveloped by example (ii), a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications. All of the other changes proposed herein are not enveloped by any specific example.

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

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: October 22, 1998.

Description of amendment request: The licensee is proposing to change Technical Specifications 3.3.2.1, ''Instrumentation—Engineered Safety Feature Actuation System Instrumentation''; 3.4.9.3, ''Reactor Coolant System [RCS]—Overpressure Protection Systems''; and 3.5.3, ''Emergency Core Cooling Systems—ECCS Subsystems—Tavg < 300 [degrees] F.'' The proposed changes will allow Millstone Unit No. 2 to prevent an automatic start of any high-pressure safety injection (HPSI) pump when the shutdown cooling system (SDCS) is in operation (Mode 4 and below). An inadvertent start of an HPSI pump could result in overpressurization of the SDCS.

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:

In accordance with 10CFR50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed changes and has concluded that they do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The proposed changes to Technical Specifications 3.3.2.1 and 3.5.3 will no longer require the HPSI pump, required to be operable in Mode 4, to start automatically on a Safety Injection Actuation Signal (SIAS). (The automatic SIASs on low pressurizer pressure and high containment pressure are not required to be operable in Mode 4. However, the manual safety injection pushbuttons are required in Mode 4). This will allow the operable HPSI pump control switch to be placed in the pull-to-lock position without affecting the operability of that pump. All HPSI pumps will be prevented from automatically starting when the plant is in Mode 4, and the Shutdown Cooling System (SDCS) is aligned to the RCS to prevent an inadvertent start of a[n] HPSI pump which could overpressurize the SDCS. These changes will not reduce the requirement for at least one HPSI pump to be operable in Mode 4. The changes will require an additional operator action to remove the operable HPSI pump breaker control switch from the pull-to-lock position, in addition to initiating safety injection by use of the manual pushbuttons, if Safety Injection System actuation is needed in Mode 4. The requirement to manually initiate a[n] HPSI pump, in addition to manually initiating a[n] SIAS, does not involve complicated equipment manipulations nor require extensive time for performing the required operator actions. The HPSI pump control switches are located in the Control Room on the same panels as the manual SIAS pushbuttons. The additional step required to start a[n] HPSI pump will not add any appreciable time for initiating HPSI flow while in Mode 4. In addition, considering the lower probability of a significant loss of coolant accident in Mode 4, and the slower plant response to a loss of coolant accident in Mode 4, the time required for the additional operator action will have no significant effect on the consequences of the accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to Technical Specification 3.4.9.3, Surveillance Requirement (SR) 4.4.9.3.3, will allow the use of the new pull-to-lock feature of the HPSI pump control switches to satisfy low temperature overpressure protection mass input requirements. This will not affect either the LTOP [low-temperature overpressure protection] HPSI pump mass input restrictions or the level of control to ensure the HPSI pumps are not capable of injecting into the RCS. The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

The proposed minor editorial and non-technical changes to add amendment numbers to Page 3/4 3–12 and to revise the wording of SRs 4.4.9.3.2 and 4.4.9.3.3 will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Bases reflect the proposed changes to the applicable Technical Specifications. The proposed changes will have no adverse effect on plant operation. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will allow the use of the HPSI pump breaker control switch pull-to-lock feature. Operation of the HPSI pump in Mode 4 will change since the operator will have to start the HPSI pump, in addition to manually initiating safety injection. However, HPSI pump operation is not an accident initiator. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification changes will no longer require the HPSI pump, required to be operable in Mode 4, to start automatically on a[n] SIAS, will allow the use of the new pull-to-lock feature of the HPSI pump control switches to satisfy low temperature overpressure protection mass input requirements, and will make minor editorial and non-technical changes. These changes will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

The only adverse impact of the proposed changes is that an additional operator action will be necessary to initiate HPSI flow in Mode 4, if needed. However, considering the lower probability of a significant loss of coolant accident in Mode 4, and the slower plant response to a loss of coolant accident in Mode 4, the time required for the additional operator action will have no significant effect on the consequences of the accident. Therefore, based on the responses above, the proposed changes are deemed safe.

The NRC has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC. The minor editorial and non-technical changes proposed herein to add page amendment numbers and clarify wording are enveloped by example (i), a purely administrative change to Technical Specifications. All of the other changes proposed herein are not enveloped by any specific example.

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

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: October 30, 1998.

Description of amendment request: Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) Surveillance Requirements 4.8.4.3.b.1, 4.8.4.3.b.2, and 4.8.4.3.b.3 list the Overvoltage (OV), Undervoltage (UV), and Underfrequency (UF) values for the protective instrumentation for the RPS electric power monitoring channels. The proposed changes correct a discrepancy between the General Electric Nuclear Engineering (GENE) Design Specification for Power Supply Monitoring Relays and the existing TS Allowable Values (AVs). The changes will revise the OV, US, and UF values from 132VAC, 109VAC, and 57Hz to 127.6VAC, 110.7VAC, and 57.05Hz respectively.

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

The proposed Tech Spec changes to section 4.8.4.3.b for the Overvoltage (OV), Undervoltage (UV), and Underfrequency (UF) relays are more conservative than the existing TS values. This change provides more protection for the associated RPS components, thus decreasing the probability of a failure in RPS. The associated Non-Conformance Report and calculation provide assurance that the OV/UV/UF settings are acceptable since the calculated values assure that the RPS components will operate within their ratings. There are no physical changes to the associated protective relays by the TS change; thus, original design basis redundancy and separation is maintained. There is no change in the interface of the RPS and its power supplies.

The safety function of the RPS is to initiate a reactor scram in order to protect the primary fission products barrier, the reactor fuel. The proposed TS Change to impose more conservative Allowable Values for the OV, UV, and UF relays will provide additional assurance that the RPS will operate within equipment voltage and frequency ratings, and will not be damaged by power system anomalies. This change will not affect the scram function of RPS; thus, the consequences of any design basis events will not be affected.

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

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

The proposed TS Allowable Values changes will not result in any physical changes to the RPS Electric Power Monitoring System. Existing setpoints will not be changed, only the TS Allowable Values are being modified to be more conservative.

The system redundancy and independence are not changed, and no new failure modes are introduced.

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

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

Currently, there are no TS bases for the existing RPS Electric Power Monitoring System OV, UV, and UF allowable values. Specific analytical limits for system voltage and frequency are not defined in the Safety Analysis Report, nor discussed in any design basis Allowed Outage Time or accident evaluation.

Investigation into the licensing basis has identified nominal values of +/−10% of 120 VAC and −5% of 60 HZ for the Allowable Values. These values are included in NUREG 0123, from which LGS's TSs were developed. NUREG 0123 also provides no bases for these values.

The proposed changes in the TS Allowable Values is based on a revision to the calculation for RPS Breaker Panel—RPS / UPS [uninterruptible power supply] System Bus Relay Settings. This revision determines the new allowable values based on the design ratings of RPS components, and factors in instrument inaccuracies and margin. These changes will also provide bases for the associated TS section. The proposed changes bring TSs into agreement with plant design specifications.

Therefore, the proposed TS 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.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: October 22, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.8.2.1.b.3 to increase the minimum battery electrolyte temperature limit from 60°F to 72°F. This change resolves a discrepancy in the electrolyte temperature assumed in the Class 1-E battery sizing calculations versus the limit specified in the TSs.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC). The Class-1E batteries will continue to function as designed. The Class-1E battery system is designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. The proposed TS surveillance testing and monitoring requirements will continue to ensure that the Class-1E batteries are capable of performing their required safety functions. In addition, this proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety, since the manner i[n] which the Class-1E battery system is operated is not affected by these proposed changes. The proposed changes merely establish TS surveillance acceptance criteria that more appropriately reflect the actual plant design. Therefore, the proposed TS changes would not result in an increase of the consequences of an accident previously evaluated.

Therefore, the proposed TS change does not involve an 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 TS changes do not involve any physical changes to the design of plant systems, structures or components. The design and operation of the Class-1E battery system is not changed from that currently described in the [Updated Final Safety Analysis Report] UFSAR, only the allocation of battery capacity design margin is affected by the increased TS minimum battery electrolyte temperature limit. The Class-1E battery system will continue to function as designed to mitigate the consequences of an accident. Implementing new TS surveillance acceptance criteria that more appropriately reflect the actual plant design does not permit plant operation in a configuration that would create a different type of malfunction to the Class-1E batteries than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in the UFSAR regarding the safety related functions of the Class-1E batteries or their support systems.

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

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

The proposed TS change involves the implementation of new TS surveillance acceptance criteria that more appropriately reflect the actual plant design. The new TS minimum battery electrolyte temperature limit enables the Class-1E battery capacity margin to be allocated in a manner which conforms to Hope Creek's current licensing basis. The ability of the Class-1E batteries to independently supply their required loads for four hours without support from battery chargers is not affected by these proposed changes. The safety-related Class-1E support systems will ensure that the proposed TS minimum electrolyte temperature limit is met.

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

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

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

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

NRC Project Director: Robert A. Capra.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: October 15, 1998, as supplemented by letter dated November 11, 1998.

Description of amendment request: The proposed amendments would change the Vogtle Electric Generating Plant, Unit 1 and Unit 2 Facility Operating Licenses to delete or modify certain license conditions, which have become obsolete or inappropriate. In addition, the Technical Specifications would be reconstituted to reflect revised word processing. No change in technical requirements would be involved; however, the font would be changed to Arial 11 point; page numbers would be revised to a limiting condition for operation specific numbering scheme; and intentional blank pages would be deleted.

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

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

The proposed changes either remove or modify provisions in the VEGP [Vogtle Electric Generating Plant] Unit 1 and [Unit] 2 Operating Licenses that have been completed or are otherwise obsolete. Each proposed change is summarized below:

Certain Surveillance Requirements (SRs) that were either added or modified at the time of Improved Technical Specifications (ITS) implementation were listed in the Operating Licenses with a schedule for performance. With the exception of Unit 2 SR 3.8.1.20, all SRs are deleted from the Operating Licenses, because they have since been performed according to schedule, and will henceforth be performed in accordance with the Technical Specifications.

A condition concerning changes to the Unit 1 initial test program is deleted due to the completion of the program.

A condition related to FEMA [Federal Emergency Management Agency] procedures and the emergency plan is deleted from the Unit 1 license due to the obsolescence of the condition.

Conditions requiring the submission of Unit 1 reports concerning the steam generator tube rupture analysis, the reactor vessel level instrumentation system, the safety parameter display system, the detailed control room design review, and the zinc coating of the diesel fuel storage tanks are deleted due to completion of the required activities.

A condition requiring modification of the Unit 1 ventilation exhaust of the alternate radwaste facility is deleted due to completion of the required activity.

An exemption related to the seismic adequacy of the Unit 1 spent fuel racks is deleted because the required actions are completed and the exemption has been determined to be no longer in effect.

A condition in both the Unit 1 and Unit 2 licenses containing reporting requirements for other license conditions is revised due to ambiguities between the requirements in the license condition and those published in NRC regulations.

A schedular exemption for the Unit 2 decommissioning funding report is deleted because the report was submitted as required and the exemption is no longer in effect.

The Technical Specifications and associated Bases have been converted from WordPerfect® for DOS version 5.1 to Microsoft® Word 97. There were no changes to technical requirements. The only visible changes to the document are as follows: (1) the font was changed to Arial 11 point; [(2)] page numbers were revised to an LCO [limiting condition for operation] specific numbering scheme; and [(3)] intentionally blank pages were deleted.

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

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

The proposed administrative/editorial changes do not alter the operation of any plant system or equipment and do not introduce a new mode of operation. Each requirement contained in the license conditions proposed for deletion has either been completed or is obsolete. Since these parts of the license are no longer applicable, deletion of these items does not provide the potential for an accident to be created. The conversion of the Technical Specifications from one word processing format to another did not involve any changes to technical requirements. Thus, the proposed changes cannot create a new accident initiating mechanism, and do not create the possibility of a new and different type of accident from any previously evaluated.

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

The license conditions proposed for deletion are obsolete and each requirement has been completed. The conversion of the Technical Specifications from one word processing format to another did not involve any changes to technical requirements. Since the proposed changes are strictly administrative/editorial and do not involve any physical or procedural changes to the plant, the margin of safety, as defined in the bases for any Technical Specification is not affected by the proposed changes.

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

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia.

NRC Project Director: Herbert N. Berkow.

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

Date of application for amendments: November 16, 1996 (TS 98–06).

Brief description of amendments: The proposed amendments would change the Sequoyah Nuclear Plant Technical Specifications (TSs) by revising the emergency diesel generator (EDG) surveillance requirements (SRs) to add a note that allows the SR to be performed in Modes 1, 2, 3 or 4, if the associated components are already out-of-service for testing or maintenance and to remove the SR that verifies certain lockout features prevent EDG starting.

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

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the TSs, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The probability of occurrence or the consequences for an accident or malfunction of equipment is not increased by this request. The proposal does not alter the way any structure, system or component functions, does not modify the manner in which the plant is operated, and does not alter equipment out-of-service time. This request does not degrade the ability of the D/G [emergency diesel generator] or equipment downstream of the load sequencers to perform their intended function. Deleting the surveillance of a nonsafety-related equipment protection function from TS likewise does not change the probability or consequences of analyzed accident scenarios. Dose consequences remain unchanged by this request.

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

A possibility for an accident or malfunction of a different type than any evaluated previously in SQN's FSAR [Final Safety Analysis Report] is not created; nor is the possibility for an accident or malfunction of a different type. The proposal does not alter the way any structure, system or component functions and does not modify the manner in which the plant is operated.

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

The margin of safety has not been reduced since the test methodologies are not being changed and LCO [Limiting Condition for Operation] allowed outage times are not being changed. Deleting the surveillance of a nonsafety-related equipment protection function from TS likewise does not reduce the margin of safety. The results of accident analysis remain unchanged by this request.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: October 27, 1998.

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. The change would apply additional conservatism by modifying the MCPR Safety Limit values, as calculated by General Electric, by maintaining the limit of 1.09 for two recirculation loop operation and by increasing the limit from 1.10 to 1.11 for single loop operation.

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

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

There is no change to any plant equipment. Per USAR Section 4.2.1, the fuel system design bases are provided in General Electric Standard Application for Reactor Fuel (GESTAR II). The Minimum Critical Power Ratio (MCPR) Safety Limit protects the fuel in accordance with the design basis. The MCPR Safety Limit calculations limit the bundle power to ensure the critical power ratio remains unchanged. Therefore, there is not an increase in the probability 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 if the limit is not violated. Therefore, there is no increase in the probability of the occurrence of a previously analyzed accident.

The fundamental sequences of accidents and transients have not been altered. The MCPR Operating Limits are selected such that potentially limiting plant transients and accidents prevent the MCPR from decreasing below the MCPR Safety Limit anytime during the transient. Therefore, there is no impact on any of the limiting USAR Appendix 15B transients. The radiological consequences are the same as previously stated in the USAR, and as approved in the NRC Safety Evaluation for GESTAR II. 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 values are designed to ensure that fuel damage from transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the limiting postulated accident. The values are calculated in accordance with GESTAR II and the fuel vendor's interim implementing procedures, which incorporate cycle-specific parameters.

The GESTAR II analysis has been accepted by the NRC as comprehensive for ensuring that fuel designs will perform within acceptable bounds. The MCPR Safety Limit ensures that the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. In addition, the initiating sequence of events has not changed. Therefore, no new or different kind of accident is created.

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

The MCPR Safety Limit values do not alter the design or function of any plant system, including the fuel. The new MCPR Safety Limit values were calculated using NRC-approved methods described in GESTAR II and the fuel vendor's interim implementing procedures, which incorporate cycle-specific parameters. The MCPR Safety Limit values are consistent with GESTAR II, the NRC Safety Evaluation of GESTAR II, the NRC Safety Evaluation Report for the Perry Nuclear Power Plant and its Supplements for USAR Sections 4.4.1 and 15.0.3.3.1, and the Technical Specification Bases (Section 2.1.1.2) for the MCPR Safety Limit. This change incorporates a cycle-specific MCPR Safety Limit, as opposed to relying on the generic limit. Therefore, the implementation of the proposed change to 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.

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

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

NRC Project Director: Stuart A. Richards.

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

Date of application request: October 27, 1998 (supersedes the April 12, 1996, amendment request). This notice supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that was published on May 8, 1996 (61 FR 20858).

Description of amendment request: The proposed amendment application would change the technical specifications (TS) for the reactor coolant system and associated Bases to allow the installation of electrosleeves in the Callaway steam generators for two fuel cycles.

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 electrosleeve configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. The applied stresses and fatigue usage for the sleeve are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Mechanical testing has shown that the structural strength of nickel electrosleeves under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by RG [Regulatory Guide] 1.121. Leakage testing for ⅝″, ⅞″, 1¹⁄₁₆″ and ¾″ tube sleeves has demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition.

The sleeve nominal wall thickness (used for developing the depth-based plugging limit for the sleeve) is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. The limiting requirement of Regulatory Guide 1.121, which applies to part throughwall degradation, is that the minimum acceptable wall must maintain a factor of safety of three against tube failure under normal operating (design) conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design condition requirement minimum wall thickness.

A bounding tube wall degradation growth rate per cycle and a NDE [Non-Destructive Examination] uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent is determined by NDE. The degradation which would require plugging sleeved tubes is developed using the guidance of RG 1.121 and is defined in BAW–10219P, to be 20% throughwall for any service induced degradation.

The consequences of failure of the sleeve are bounded by the current steam generator tube rupture analysis included in the Callaway FSAR [Final Safety Analysis Report]. 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 risk assessment for installation of Electrosleeves at Callaway Plant was performed for a two-cycle operating period. The results of this evaluation determined that sufficient margins against postulated tube rupture during bounding accident conditions exist for all types of degradation of the Electrosleeve material. The calculated probability of burst for a hypothetical population of 10,000 axial flaws, 100% throughwall of the parent tube and 0.40″ long, is 4.4×10^{-11} at the end of the second operating cycle. The probability of burst for postulated circumferential flaws and pits is determined to be essentially zero.

The proposed change does not adversely impact any other previously evaluated design basis accident or the results of LOCA [Loss of Coolant Accident] and non-LOCA accident analyses for the current technical specification minimum reactor coolant system flow rate. The results of the analyses and testing demonstrate that the electrosleeve is an acceptable means of maintaining tube integrity. Furthermore, per Regulatory Guide 1.83 recommendations, the sleeved tube can be monitored through periodic inspections with present NDE techniques. These measures demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Conformance of the electrosleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests, support the conclusion that installation of electrosleeves will 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.

Electrosleeving does not represent a potential to adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of electrosleeving maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. Leak and mechanical testing of electrosleeves support the conclusions of the calculations that each sleeve retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. 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.

Implementation of sleeving will reduce the potential for primary to secondary leakage during a postulated steam line break while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, the potential for steam line break leakage is reduced. These degraded intersections now are returned to a condition consistent with the Design Basis. While the installation of a sleeve reduces primary coolant flow, the reduction is far below that caused by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving versus plugging.

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

The electrosleeve repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition, i.e., tube/sleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Code used in steam generator design. The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in BAW–10219P.

In addition, since the installed sleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed. The effect of sleeving on the design transients and accident analyses has been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate and the Callaway Safety Analysis.

Provisional requirements cited in other NRC Safety Evaluation Reports addressing the implementation of sleeving have required the reduction of the individual steam generator normal operation primary to secondary leakage limit from 500 to 150 gpd [gallons per day]. Consistent with these evaluations, Union Electric will reduce the per steam generator leak rate of 500 gpd in TS 3.4.6.2.c to 150 gpd. The establishment of this leakage limit at 150 gpd provides additional safety margin. [The staff notes that this leakage limit has been incorporated into the Callaway Technical Specifications via license amendment #119 dated October 1, 1996.]

Finally, Union Electric will reduce the tube plugging limit from 48% through wall to 40% through wall to be consistent with NUREG–1431. The establishment of the plugging limit at 40% through wall provides additional safety margin. [The staff notes that this plugging limit has been incorporated into the Callaway Technical Specifications via license amendment #119 dated October 1, 1996.]

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

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of amendment request: November 3, 1998.

Description of amendment request: The licensee proposes to make administrative changes to the Technical Specifications to correct errors, add consistency within the Technical Specifications, and make nomenclature changes to support and enhance usability of the Technical Specifications.

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

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

The proposed changes are purely administrative in nature and have no effect on plant hardware, plant design, safety limit setting, or plant system operation and therefore do not modify or add any initiating parameters that would significantly increase the probability or consequences of an accident previously evaluated.

No new modes of operation are introduced by the proposed changes such that adverse consequences would result. Accordingly, the consequences of previously analyzed accidents are not affected by this proposed license amendment.

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

These changes do not affect the operation of any systems or components, nor do they involve any potential initiating events that would create any new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated for the Vermont Yankee Nuclear Power Station.

3. Involve a significant reduction in a margin of safety, because:

These proposed changes do not affect any equipment involved in potential initiating events or safety limits. Therefore, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

Administrative changes, as such, do not constitute any significant hazards considerations.

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

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

NRC Project Director: Cecil O. Thomas.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: November 10, 1998.

Description of amendment request: The proposed changes to North Anna Power Station (NAPS), Units 1 and 2, Technical Specification (TS) 3.4.4 will clarify the operability requirements for the pressurizer heaters and eliminate a potential verbatim compliance issue associated with the pressurizer heaters and emergency power supply. The verbatim compliance issue was created when the Emergency Diesel Generator allowed outage time was changed from 72 hours to 14 days.

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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed changes will revise the LCO [limiting condition for operation] 3.4.4 to require that the pressurizer have two groups of pressurizer heaters operable with a capacity of greater than or equal to 125 kW and capable of being powered from its associated emergency bus. The Action Statement will also be revised to focus on heater operability. The following is provided to support this conclusion.

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

The pressurizer heaters are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not increased. The pressurizer heaters remain operable as assumed in the accident analysis to mitigate the consequences of any accident. Therefore, the proposed changes to clarify the operability requirements do not significantly increase the probability of occurrence or the consequences of any previously analyzed accident.

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

The proposed Technical Specifications changes do not involve any physical alteration of the plant or changes in methods governing normal plant operation. Operation of and the design of the pressurizer heaters and the associated power supplies are not changed by the proposed changes. The proposed changes do not impose any new or eliminate any existing requirements. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

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

The proposed Technical Specifications changes will not reduce the margin of safety since the change has no effect on any safety analyses assumptions. The pressurizer heaters remain operable as assumed in the safety analysis to mitigate the consequences of any accident previously analyzed. The proposed changes only clarify the operability requirements for the pressurizer heaters and associated emergency power supplies. Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

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

NRC Project Director: Herbert N. Berkow.

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

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

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

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

Date of amendment request: October 29, 1998.

Description of amendment request: Technical Specification changes (TS) relating to the implementation and automatic removal of certain reactor protection system trip bypasses to ensure that the meaning of explicit terms used in the TSs are consistent with the intent of the stated requirements.

Date of publication of individual notice in the **Federal Register**: November 5, 1998 (63 FR 59809).

Expiration date of individual notice: November 19, 1998.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Baltimore Gas and Electric Company, Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of application for amendment: July 20, 1998.

Brief description of amendment: The amendment implements a modification that constitutes an unreviewed safety question as described in 10 CFR 50.59. The modification involves replacing the service water heat exchangers with new plate and frame heat exchangers having an increased thermal performance capability. The planned modification is similar to the one completed on Unit 1. In addition, by a separate letter dated July 20, 1998, the licensee submitted a request to obtain approval for a temporary one time cooling lineup needed to support emergency diesel generator operability for the installation of the Unit 2 service water heat exchanger replacement, which is currently being reviewed by the NRC staff. Therefore, since the implementation of the proposed service water heat exchanger modification is dependent on the staff's issuance of the one time Technical Specification (TS) change regarding installation of the modification, this modification should not be implemented prior to the issuance of the one-time TS change for installing the modification.

Date of issuance: November 5, 1998.

Effective date: This license amendment is effective as of the date of its issuance to be implemented after the staff's issuance of the one-time TS change regarding the installation of the service water heat exchanger modification.

Amendment No.: 203.

Facility Operating License No. DPR–69: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: August 12, 1998 (63 FR 43201).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 1998.

No significant hazards consideration comments received: No.

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

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: June 26, 1998.

Brief description of amendment: The amendment modifies various Technical Specification pages to correct typographical errors, remove inadvertent replication of information, and updates various Bases sections.

Date of issuance: November 10, 1998.

Effective date: November 10, 1998.

Amendment No: 178.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50933).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 10, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

Date of application for amendment: March 6, 1998, as supplemented September 11, 1998. The September 11, 1998, supplemental letter contained clarifying information only, and did not change the no significant hazards consideration determination.

Brief description of amendment: The amendment revises Technical Specification 3.9.2 relating to the use of Post-Accident Monitoring Source Range neutron flux detectors as a compensatory measure in the event that one of the two required BF3 neutron flux detectors becomes inoperable during Mode 6 operations (refueling).

Date of issuance: November 12, 1998.

Effective date: November 12, 1998.

Amendment No: 180.

Facility Operating License No. DPR–23: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: June 3, 1998 (63 FR 30262).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1998.

No significant hazards consideration comments received: No.

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

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 17, 1998, as supplemented October 15, 1998.

Brief description of amendments: The amendments revised the Updated Final Safety Analysis Report to perform a Keowee Emergency Power Engineered Safeguards Functional Test during the 1998 Unit 3 refueling outage at Oconee.

Date of Issuance: November 12, 1998.

Effective date: As of the date of issuance to be implemented during the 1998 Unit 3 refueling outage.

Amendment Nos.: Unit 1—233; Unit 2—233; Unit 3—232.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: September 30, 1998 (63 FR 52304).

The October 15, 1998, letter provided clarifying information that did not change the scope of the September 17, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Dauphin County, Pennsylvania

Date of application for amendment: December 2, 1996.

Brief description of amendment: This amendment would revise audit frequency requirements and relocate them from the Technical Specifications to the Quality Assurance Plan.

Date of issuance: November 12, 1998.

Effective date: This amendment is effective immediately to be implemented written 60 days.

Amendment No.: 52.

Facility Operating License No. DPR–73: The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40850).

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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

Date of application for amendments: May 11, 1998, as supplemented by letter dated October 9, 1998.

Brief description of amendments: The amendments modify the technical specifications (TS) for San Onofre Nuclear Generating Station Unit Nos. 2 and 3 to implement 10 CFR Part 50 Appendix J, Option B for performance-based reactor containment leakage testing.

Date of issuance: November 6, 1998.

Effective date: November 6, 1998, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2 –144; Unit 3 –135.

Facility Operating License No. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48265).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

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

Date of amendment request: July 6, 1998.

Brief description of amendments: Relocates the description of the reactor coolant system design features in Technical Specification 5.4 to the Updated Final Safety Analysis Report, which already contains the information.

Date of issuance: November 18, 1998.

Effective date: November 18, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 98; Unit 2—Amendment No. 85.

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

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48266).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 6, 1998 , as supplemented on October 28, 1998.

Brief description of amendments: Relocate the Technical Specification 3/4.3.3.3 requirements for Seismic Instrumentation to the Technical Requirements Manual.

Date of issuance: November 18, 1998.

Effective date: November 18, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 99; Unit 2—Amendment No. 86.

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

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48267).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 6, 1998, as supplemented on October 28, 1998.

Brief description of amendments: Relocates the Technical Specification 3/4.7.13 requirements for the Area Temperature Monitoring System to the Technical Requirements Manual.

Date of issuance: November 18, 1998.

Effective date: November 18, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 100; Unit 2—Amendment No. 87.

Facility Operating License Nos. NPF–76 and NPF–80: The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48267). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 13, 1998 (TS 97–07).

Brief description of amendments: The amendments incorporate new main steam isolation valve (MSIV) requirements that are consistent with NUREG–1431, the Westinghouse Standard Technical Specifications (TS), including testing requirements for the MSIVs that ensure the valves close on an automatic actuation signal.

Date of issuance: November 17, 1998.

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

Amendment Nos.: 236 and 226.

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

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19980).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 1998.

No significant hazards consideration comments received: No.

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

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: June 26, 1998 (TS 98–02).

Brief description of amendments: The amendments change the Technical Specifications and their Bases to lower the specific activity limit for the primary coolant system from 1.0 microcurie/gram dose equivalent iodine-131 to 0.35 microcurie/gram, as provided for in NRC Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.'' This change allows a proportional increase in main steam line break induced primary-to-secondary leakage when implementing the alternate steam generator tube repair criteria, which the NRC has already approved for Sequoyah Nuclear Plant, Units 1 and 2.

Date of issuance: November 17, 1998.

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

Amendment Nos.: 237 and 227.

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

Date of initial notice in **Federal Register**: July 15, 1998 (63 FR 38205).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 5, 1998 (TS 98–008).

Brief description of amendment: This amendment is in response to your application dated August 5, 1998. The amendment revises the WBN Technical Specifications (TS) and associated TS Bases to allow up to 4 hours to make the residual heat removal suction relief valve available as a cold overpressure mitigation system relief path.

Date of issuance: November 10, 1998.

Effective date: November 10, 1998.

Amendment No.: 14.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: September 23, 1998 (63 FR 50940).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 10, 1998.

No significant hazards consideration comments received: None.

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

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

Date of amendment request: July 10, 1996 (TXX–96405), as supplemented by letters dated October 1, 1996 (TXX–96475), and July 1, 1998 (TXX–98159).

Brief description of amendments: The amendment would take credit for the addition of train oriented Fan Coil Units for each UPS and Distribution Room and would provide redundancy to the existing Air Conditioning (A/C) Units (TS 3/4.7.11 and its associated bases).

Date of Issuance: Date of issuance: November 18, 1998.

Effective date: November 18, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 61; Unit 2—Amendment No. 47.

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

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6579).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: May 7, 1998.

Brief description of amendment: This amendment revises Technical Specification 5.4, ''Fuel Storage,'' to increase the allowable mass of uranium-235 (U^{235}) per axial centimeter for fuel storage. The requested change will allow the use of new Siemens Power Corporation heavy fuel assembly designs.

Date of Issuance: November 12, 1998.

Effective date: November 12, 1998.

Amendment No.: 141.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33111).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

Dated at Rockville, Maryland, this 24th day of November 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–31931 Filed 12–1–98; 8:45 am]

BILLING CODE 7590–01–P

PRESIDIO TRUST

Management of the Presidio

AGENCY: The Presidio Trust.

ACTION: Notice of availability for public comment.

SUMMARY: This notice announces the availability of and requests comments on the Interim Compendium compiled pursuant to final interim regulations concerning management of the area under the administrative jurisdiction of the Presidio Trust. The final interim regulations were adopted by the Presidio Trust as 36 CFR parts 1001, 1002, 1004, and 1005 and published in the **Federal Register** on June 30, 1998 (63 FR 35694).

DATES: Comments on the Interim Compendium must be received by January 29, 1999.

ADDRESSES: Written comments on the Interim Compendium must be sent to Karen A. Cook, General Counsel, The Presidio Trust, 34 Graham Street, P.O. Box 29052, San Francisco, CA 94129–0052.

FOR FURTHER INFORMATION CONTACT: Karen A. Cook, General Counsel, The Presidio Trust, 34 Graham Street, P.O. Box 29052, San Francisco, CA 94129–0052. Telephone: 415–561–5300.

SUPPLEMENTARY INFORMATION:

Background

The Presidio Trust's final interim regulations at 36 CFR parts 1001, 1002, 1004, and 1005 provide that the Board of Directors of the Presidio Trust ''shall compile in writing all the designations, closures, permit requirements and other restrictions imposed under discretionary authority.'' 36 CFR 1001.7(b). The Board has compiled these in an Interim Compendium. This Interim Compendium was approved by the Board on June 30, 1998 and is currently in effect.

Although public notice and comment on this Interim Compendium is not required by the Trust's regulations or other applicable authority, the Trust's Board has decided to make the Interim Compendium available for public comment for a period of 60 days. Following the public comment period, the Trust will consider any comments received and make any appropriate changes to the Interim Compendium. Because the Trust is currently engaged in a rulemaking concerning management of the Presidio and various administrative matters, the Trust may make other changes to the Interim Compendium both during this comment period and following its close.

How to Obtain Copies

During this comment period, a copy of the Interim Compendium is available for public inspection and copying during normal office hours (9:00 a.m. to 5:00 p.m., excluding Saturdays, Sundays, and Federal holidays) at the offices of the Presidio Trust, 34 Graham Street, The Presidio, San Francisco, CA 94129. Prior to the close of the comment period, upon receipt of a written request and advance payment by check or money order to the Presidio Trust in the amount of $2.40 for photocopying charges, the Trust will mail a copy of the Interim Compendium to any interested member of the public.