should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to Mr. John M. Fulton, Assistant General Counsel, Entergy Nuclear Generating Co., Pilgrim Station, 600 Rocky Hill Road, Plymouth, MA 02360, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the 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 dated September 6, 2000, which was submitted by the Power Authority of the State of New York and which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 18th day of January 2001.

For the Nuclear Regulatory Commission.

George F. Wunder,

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

[FR Doc. 01–2114 Filed 1–23–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 2, 2001, through January 12, 2001. The last biweekly notice was published on January 10, 2001 (66 FR 2010).

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

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

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

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

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

By February 23, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first Floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: December 29, 2000.

Description of amendment request: The proposed amendment would increase the Technical Specification allowed outage time from 3 days to 14 days for a single inoperable Division 1 or 2 diesel generator.

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 any accident previously evaluated.

The proposed Technical Specification (TS) changes revise the Completion Time for Required Actions A.2 and B.4 associated with the Division 1 and Division 2 Diesel Generators (DG). The proposed changes allow an extension of the current TS Completion Time from 72 hours to 14 days when the Division 1 or Division 2 DG is inoperable.

The proposed changes do not affect the design of the DGs, the operational characteristics of function of the DGs, the interfaces between the DGs and other plant systems, or the reliability of the DGs. Required Actions and the associated Completion Times are not initiating conditions for any accident previously evaluated, and the DGs are not initiators of any previously evaluated accidents. The DGs mitigate the consequences of previously evaluated accidents including a loss of offsite power. The consequences of a previously analyzed event will not be significantly affected by the extended DG Completion Time since the DGs will continue to be capable of performing their accident mitigation function as assumed in the accident analysis. Thus the consequences of accidents previously analyzed are unchanged between the existing TS requirements and the proposed changes. The consequences of an accident are independent of the time the DGs are out of service as long as adequate DG availability is assumed. The proposed changes will not result in a significant decrease in DG availability so that the assumptions regarding DG availability are not impacted.

To fully evaluate the effect of the proposed EDG Completion Time extension, Probabilistic Risk Assessment (PRA) methods and a deterministic analysis were utilized. The results of the analysis show no significant increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Therefore, the proposed changes do not involve significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed changes do not involve a change in the design, configuration, or method of operation of the plant. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. The changes do not alter assumptions made in the safety analysis. No alteration in the procedures, which ensure that the plant remains within analyzed limits, is being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since there are no changes to the plant design and safety analysis, and no changes to the DG design, including any instrument setpoints, no margin of safety assumed in the safety analysis is affected. If a margin of safety is ascribed to DG availability and plant risk, it has also been determined that such a margin of safety is not significantly reduced, as the proposed changes have been evaluated both deterministically and using a risk-informed approach. The evaluation concluded the following with respect to the proposed changes.

Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, sufficient safety margins will be maintained, and any increases in CDF and LERF are small and consistent with the NRC Safety Goal Policy Statement (**Federal Register**, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986, as interpreted by NRC Regulatory Guides 1.174 and 1.177). Furthermore, increases in risk posed by potential combinations of equipment out of service during the proposed DG extended Completions Time will be managed under a configuration risk management program consistent with 10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' paragraph (a)(4). The following are examples.

- An extended DG Completion Time will not be entered intentionally for scheduled maintenance purposes if severe weather conditions are expected.
- While in the extended DG Completion Time, additional elective equipment maintenance or testing or equipment failure will be evaluated. Activities that yield unacceptable results will be avoided.
- The condition of the offsite power supply and switchyard will be evaluated.
- Activities have been identified that can mitigate any increase in risk. Procedures are in place for the minimizing risk associated with the following activities:

No elective maintenance will be scheduled within the switchyard that would challenge the offsite power connection or offsite power availability during the extended DG Completion Time.

No elective work will be performed on protected equipment or opposite train emergency core cooling system (ECCS) equipment during the extended DG Completion Time.

The availability of offsite power coupled with the availability of the other DGs and the use of on-lime risk assessment tools provide adequate compensation for the potential small incremental increase in plant risk of the extended DG Completion Time. In addition, the increased availability of the DGs during refueling outages offsets the small increase in plant risk during operation. The proposed extended DG Completion Times in conjunction with the availability of the other DGs continues to provide adequate assurance of the capability to provide power to the engineered safety features (ESF) buses. Therefore, implementation of the proposed changes will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW, Washington, DC 20036–5869.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 6, 2000.

Description of amendment request: The proposed amendment requests changes to the once-through steam generator tube inspection criteria in order to allow certain inside diameter inter-granular attack indications to remain in service. This amendment request seeks to make permanent the tube inspection criteria that have been used for the past two operating cycles at TMI–1.

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

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

The proposed flaw disposition strategy, based on measurable eddy current parameters of axial and circumferential extent for Inside Diameter (ID) Initiated Inter-Granular Attack (IGA), will continue to provide high confidence that unacceptable flaws that do not have the required structural integrity to withstand a postulated MSLB [main steam line break] are removed from service. The axial and circumferential length limits for eddy current ID degradation indications meet Draft Regulatory Guide 1.121 (Reference 9 [of the licensee's application]) acceptance criteria for margin to failure for MSLB-applied differential pressure and axial tube loads. The capability for detection of flaws is unaffected; and the identification of tubes that should be repaired or removed from service is maintained. The operation of the OTSGs [once-through steam generators] or related structures, systems, or components is otherwise unaffected. Therefore, neither the probability nor consequences of a Steam Generator Tube Rupture (SGTR) is significantly increased either during normal operation or due to limiting loads of a MSLB accident.

Therefore, operation of the facility in accordance with the changes included in LCA [license change application] No. 291 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no hardware changes involved nor changes to any operating practices. These changes involve only the OTSG tube inservice inspection surveillance requirements, which could only affect the potential for OTSG primary-to-secondary leakage which has been analyzed and is subject to Technical Specification requirements not affected by these changes. The proposed changes continue to impose flaw length limits for ID IGA to assure tube structural and leakage integrity.

Therefore, operation of the facility in accordance with the changes included with LCA No. 291 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margins of safety defined in Draft Regulatory Guide 1.121 (Reference 9 [of the licensee's application]) are retained. The probability of detecting degradation is unchanged since the bobbin coil eddy current methods will continue to be the primary means of initial detection and the probability of leakage from any indications left in service remains acceptably small. The strategy of dispositioning ID-initiated IGA indications will continue to provide a high level of confidence that tubes exceeding the allowable limits for tube integrity are repaired or removed from service.

Therefore, operation in accordance with the changes included in LCA No. 291 will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: December 6, 2000.

Description of amendment request: The proposed amendment provides clarifications to the decay heat removal (DHR) Technical Specifications (TSs). It is intended, in part, to fulfill a commitment made by the licensee to the NRC during a pre-decisional enforcement conference on April 23, 1999. Specifically, the proposed changes would: (1) Define and clarify the emergency feedwater (EFW) flowpath redundancy as described in the Bases; (2) provide operability requirements for the redundant steam supply paths to the turbine-driven EFW pump; (3) provide a more conservative 72-hour allowed outage time (AOT) with any EFW pump or flowpath inoperable; (4) provide a more conservative 1-hour AOT with both EFW flowpaths to a single once-through steam generator (OTSG) inoperable or with 2 EFW pumps inoperable; and (5) revise and clarify EFW pump and flowpath operability requirements during surveillance testing. Minor administrative and editorial changes are also proposed. A change to the Bases for TS 3.5.5, ''Accident Monitoring Instrumentation,'' regarding the description of the pressurizer level instrument channels to reflect the replacement of Bailey transmitters was also included.

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

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

This change incorporates the concept of EFW flowpath redundancy thoughout the TS[s], which takes into consideration the redundancy provided by the EFW System modifications made in the mid-1980s after the accident at TMI–2. This change incorporates a 72 hour required action time when redundant components are made inoperable. These changes do not result in any change to the configuration of the EFW System as described in the [UF]SAR [Updated Final Safety Analysis Report] or used in plant specific analyses. The reliability of EFW System components is unaffected. The 72 hour required action time for inoperability of redundant EFW components ensures that the EFW System can fulfill its safety function to provide adequate OTSG cooling during a design basis accident (DBA). The one hour required action time ensures prompt action to initiate a plant shutdown when the design flow capability of the EFW System cannot be assured.

The current TS 4.9.1.2 contains EFW flowpath operability requirements during surveillance testing rather than requiring that a specific test be performed as do the other subparagraphs of TS 4.9.1. For this reason the requirements of TS 4.9.1.2 are being moved to the LCO [limiting condition for operation] section in Chapter 3 and combined with the note following the current TS 3.4.1.1.a(2) into a new TS 3.4.1.1.a(4) to define the EFW System operability requirements for EFW pumps and flowpaths during surveillance testing. The new specification incorporates the consideration of EFW flowpath redundancy consistent with HSPS [Heat Sink Protection System] train operability requirements and continues to require that compensatory measures be implemented to promptly restore components if EFW is needed during surveillance testing when more than one flowpath is made inoperable to an OTSG. The intent of this surveillance standard has been retained, which assures that the minimum number of EFW flowpaths to the OTSGs will be available with minimal operator action.

This change provides further assurance that EFW System design basis requirements will be met and does not affect EFW System configuration, setpoints, or reliability. These changes will not affect any accident initiation sequence and do not affect off site dose consequences of accidents that have been analyzed.

The editorial changes included in this LCA [license change application] are intended to improve the clarity, consistency, and reliability of the TS[s] [and] do not change the intent or interpretation.

Therefore, operation of the facility in accordance with the changes included in LCA–286 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As a result of this change, no additional hardware is being added; and there will be no effect on EFW System design, operation as described in the [UF]SAR, or assumptions used in plant specific analyses. The requirement for three EFW Pumps and [associated] flowpaths to be operable for continuous plant operation is not affected by this change. Events involving the EFW System operation have been reviewed and determined to have no impact from these changes. The additional operability requirements for the turbine-driven EFW Pump steam supplies, the revised LCOs [limiting condition for operation], and changes to define EFW flowpath redundancy ensures minimum EFW component operability as credited in plant analyses. The editorial changes included in this LCA are intended to improve clarity, consistency and readability of the TS[s] and Bases, [and] do not change the intent or interpretation.

Therefore, operation of the facility in accordance with the changes included with LCA–286 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the EFW System design or instrumentation setpoints. The requirement for three operable EFW pumps and associated flowpaths is not affected by this change. The revised LCO imposes a 72 hour required action time when any EFW pump or redundant flowpath to either OTSG is inoperable, including inoperability for the purpose of conducting surveillance testing. The revised LCO requires that at least one flowpath to each OTSG must be operable or a plant shutdown is required to be initiated within one hour. The 8 hour action time currently allowed for pump inoperability during surveillance testing is also applied to flowpath inoperability during testing. The revised LCO continues to require compensatory measures during EFW testing when HSPS [heat sink protection system] is required to be operable and an OTSG is isolated, retaining the provision that EFW flowpath valves can be realligned promptly from their test mode to their operational allignment if EFW flow is needed. The revised Accident Monitoring Instrumentation specification is needed to reflect the revised flowpath definition and does not change the intent of the specification. The editorial changes included in this LCA are intended to improve the clarity, consistency, and readability of the TS[s] [and] do not change the intent or interpretation.

Therefore, operation in accordance with the changes included in LCA–286 will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendments request: December 1, 2000.

Description of amendments request: The proposed amendments would revise the value of the minimum departure from nucleate boiling ratio (DNBR) from ''≥ 1.30'' in the current technical specifications to ''≥ 1.3 (through operating cycle 10)'' and ''≥ 1.34 (operating cycle 11 and later)'' in the safety limits Technical Specification (TS) 2.1.1.1 and in function 15, DNBR—Low, in Table 3.3.1–1, ''Reactor Protective System Instrumentation.'' The proposed amendments are structured such that the ''≥ 1.34'' would become effective for each unit in operating cycle 11 and later. Operating cycle 11 begins in spring 2002 for Unit

2, in fall 2002 for Unit 1, and in spring 2003 for Unit 3. From now to operating cycle 11, the ''≥ 1.30'' will remain the minimum DNBR requirement for the three units.

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

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

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The purpose of the proposed Technical Specification (TS) amendment is to provide a revised Departure from Nucleate Boiling Ratio (DNBR) Safety Limit (TS Section 2.1.1.1) and Low DNBR Reactor Protective System (RPS) trip setpoint (TS Limiting Condition for Operation (LCO) 3.3.1, Table 3.3.1–1).

The proposed TS amendment involves increasing the DNBR Safety Limit and Low DNBR RPS trip setpoint from ''≥ 1.30'' to ''≥ 1.34''. Changing this limit in and of itself will not alter the physical characteristics of any component involved in the initiation of an accident. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The Core Operating Limit Supervisory System (COLSS) Power Operating Limit (POL) is an alarm limit on the maximum steady state core power level. The alarm is based on maintaining COLSS calculated DNBR a pre-determined amount above the DNBR Safety Limit. The Low DNBR RPS trip setpoint[,] in conjunction with the COLSS POL, prevents the DNBR in the limiting coolant channel in the core from violating the DNBR Safety Limit during design basis Anticipated Operational Occurrences (AOO). Operating below the COLSS POL ensures the Low DNBR RPS trip setpoint will protect the core [fuel] from damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the worst AOO. Thus, during normal and anticipated operation the Low DNBR RPS trip setpoint in conjunction with the COLSS POL prevents overheating of the fuel cladding and subsequent cladding perforation that would release fission products to the reactor coolant.

This change will accommodate increased DNBR sensitivity to uncertainties in inlet flow to the hot assembly and adjacent assemblies. This increased sensitivity is attributed to the flatter power distributions of the more efficient present day erbium core designs. More adverse DNBR sensitivity to inlet flow was first encountered in Unit 1 Cycle 7. At that time the increased DNBR sensitivity was accounted for statistically by applying a thermal margin penalty to Core Operating Limit Supervisory System (COLSS) and Core Protection Calculators (CPCs) using approved Statistical Combination of Uncertainties (SCU) methods. This approach was also used for the subsequent cycles in all units up until the present. The NRC Safety Evaluation (issued May 26, 1994 for Palo Verde Nuclear Generating Stations (PVNGS) Units 1, 2, and 3) for the present ''≥ 1.30'' DNBR limit states, ''Uncertainties in inlet flow to the hot assembly and adjacent assemblies can be accounted for statistically by either increasing DNBR or applying a thermal margin penalty using approved SCU methods.''

The proposed TS amendment change for DNBR Safety Limit and Low DNBR RPS trip setpoint limit (≥ 1.34) was calculated using approved SCU methods to statistically include the above described increased DNBR sensitivity. This new DNBR limit was calculated such that it has a high probability of covering all future cycle designs. Thus, this change involves moving the existing increased inlet flow uncertainty penalty from a thermal margin penalty contained within COLSS and CPCs to an increase in the DNBR Safety Limit and Low DNBR RPS trip setpoint limit. The DNBR Safety Limit and Low DNBR RPS trip setpoint increases from ''≥ 1.30'' to ''≥ 1.34'' due to this change. The COLSS and CPCs would respond similarly with the increased inlet flow uncertainty penalty located in either the COLSS or CPCs or in the DNBR Safety Limit. The proposed amendment changes only the location of the increased inlet flow uncertainty penalty and does not impact the operation of the plant. The core power distribution during all phases of normal and anticipated operational occurrences will remain bounded by the initial conditions assumed in Chapter 15 of the Palo Verde Nuclear Generating Station (PVNGS) UFSAR [Updated Final Safety Analysis Report]. Thus, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This change does not alter the physical design of any System, Structure, or Component (SSC) of the plant.

The change involves increasing the DNBR Safety Limit and the Low DNBR RPS trip setpoint from ''≥ 1.30'' to ''≥ 1.34'' and decreasing the corresponding DNBR thermal margin penalty factors in COLSS and CPC in a compensating manner. Changing these limits and penalty factors will not alter the physical or functional characteristics of any component in the plant. These changes will not affect any safety-related equipment used in the mitigation of anticipated operational occurrences or design basis accidents. Thus, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The DNBR Safety Limit specified in Section 2.1.1.1 and the Low DNBR RPS trip setpoint specified in Table 3.3.1–1 of LCO 3.3.1 of [the] PVNGS Technical Specifications ensure that operation of the reactor does not result in a departure from nucleate boiling during normal operation and design basis anticipated operational occurrences. Therefore, operating consistent with the increased DNBR Safety Limit and Low DNBR RPS trip setpoint will ensure that no anticipated operational occurrences will result in core conditions below the specified DNBR Safety Limit and no postulated accident exceeds the site boundary dose limits. The UFSAR Chapter 15 analysis remains bounding and the margins of safety will be maintained because the COLSS and the CPC overall uncertainty factors will be calculated and implemented consistent with the increased DNBR Safety Limit of ''≥ 1.34''. Therefore, this change to TS Section 2.1.1.1 and Table 3.3.1–1 of LCO 3.3.1 does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: December 5, 2000.

Description of amendments request: The proposed amendments would revise the action statement for Specification 3.7.5, ''Auxiliary Feedwater (AFW) System,'' of the Technical Specifications (TSs). The amendments would incorporate NRC-approved TS Task Force (TSTF) Traveler Number TSTF–340, Revision 3, to allow a 7-day Completion Time for the turbine-driven AFW pump if inoperability occurs in reactor Mode 3 following a refueling outage, and if Mode 2 had not been entered.

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

1. Does the proposed change involve a significant increase in the probability or

consequences of an accident previously evaluated?

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment to Technical Specification 3.7.5 would allow a 7 day Completion Time for Condition A for the turbine-driven Auxiliary Feedwater (AFW) pump if the inoperability occurs in MODE 3 following a refueling outage, if MODE 2 had not been entered. Extending the Completion Time does not involve a significant increase in the probability or consequences of an accident previously evaluated because: (1) The proposed amendment does not represent a change to the system design, (2) the proposed amendment does not prevent the safety function of the AFW system from being performed since the other fully redundant esstential train and the non-essential train are required to be operable, (3) the proposed amendment does not alter, degrade, or prevent action described or assumed in any accident described in the PVNGS [Palo Verde Nuclear Generating Station] UFSAR [Updated Final Safety Analysis Report] from being performed since the other trains of AFW are required to be operable, (4) the proposed amendment does not alter any assumptions previously made in evaluating radiological consequences, and (5) the proposed amendment does not affect the integrity of any fission product barrier. No other safety related equipment is affected by the proposed change. Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment to Technical Specification 3.7.5 would allow a 7 day Completion Time for Condition A for the turbine-driven Auxiliary Feedwater (AFW) pump if the inoperability occurs in MODE 3 following a refueling outage, if MODE 2 had not been entered. Extending the Completion Time does not create the possibility of a new or different kind of accident from any accident previously evaluated because: (1) The proposed amendment does not represent a change to the system design, (2) the proposed amendment does not alter how equipment is operated or the ability of the system to deliver the required AFW flow, and (3) the proposed amendment does not affect any other safety related equipment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The PVNGS safety analysis credits essential Auxiliary Feedwater (AFW) pump delivery of 650 gpm at a steam generator pressure of 1270 psia or equivalent at the steam generator entrance for design basis accidents. The AFW System Design Basis Manual (AF), Revision 11, states that these pumps are designed to supply 750 gpm. The proposed [***] amendment to Technical Specification 3.7.5 would allow a 7 day Completion Time for Condition A for the turbine-driven AFW pump if the inoperability occurs in MODE 3 following a refueling outage, if MODE 2 had not been entered. Extending the Completion Time does not involve a significant reduction in a margin of safety because: (1) During a return to power operations following a refueling outage, decay heat [in the core] is at its lowest levels, (2) the other essential and non-essential AFW trains are required to be OPERABLE when MODE 3 is entered, (3) the essential motor-driven AFW train can provide sufficient flow to remove decay heat and cool the unit to Shutdown Cooling system entry conditions from power operations, and 4) the non-essential motor-driven AFW train is designed to supply sufficient water to remove decay heat with steam generator pressure at no load conditions to cool the unit to Shutdown Cooling entry conditions.

Based on the responses to these three criteria, APS [Arizona Public Service Company] has concluded that the proposed amendment involves no significant hazards considerations.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 13, 2000

Description of amendment request: The proposed amendment would revise Harris Nuclear Plant (HNP) Technical Specification (TS) 3/4.9.2 ''Refueling Operations—Instrumentation'' and the associated Bases to permit using alternate installed detectors or temporary source range detectors instead of the two Source Range Nuclear Flux Monitors specified in the current HNP TS.

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

This change only involves reactor core monitoring requirements during Mode 6. These monitoring requirements are not credited for accident mitigation. Alternate monitors will be provided with the accuracy and sensitivity required to adequately monitor changes in the core reactivity levels during refueling activities. Neutron Flux monitors are for indication only and do not interface with other structures, systems, or components that might initiate an accident.

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

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

Neutron Flux monitors are for indication only and do not interface with other structures, systems, or components that might initiate an accident. The proposed change will not modify plant systems or operate plant components such that a new or different accident scenario is created.

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

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

Similar changes, to the proposed change, have been approved at the Beaver Valley Power Station and the Diablo Canyon Power Plant. The proposed change will maintain adequate monitoring of core reactivity in Mode 6. The proposed change maintains requirements for two operable neutron flux monitors. Neutron flux monitors are not credited in the HNP accident analyses for accident mitigation in Mode 6.

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

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 14, 2000.

Description of amendment request: The proposed amendment would revise Harris Nuclear Plant (HNP) Technical Specification (TS) 3/4.8.1 related to emergency diesel generators (EDGs). Specifically, the licensee proposes revising TS Surveillance Requirement 4.8.1.1.2.f.7, the 24-hour EDG endurance run test, by removing the restriction to perform the test during shutdown conditions.

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

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

The EDGs and their associated emergency buses are not accident initiating equipment; therefore, there will be no impact on accident probabilities due to this proposed amendment. The EDGs mitigate the consequences of previously evaluated accidents involving a loss of offsite power. The proposed amendment continues to assure the EDGs perform their function when called upon. The design of the equipment is not being modified. The proposed amendment does not impact the operational characteristics of the EDGs, the interfaces between the EDGs and other plant systems, or the function or reliability of the EDGs. The EDGs remain capable of performing their accident mitigation function. The HNP Probabilistic Safety Analysis (PSA) model results are not affected by the proposed change.

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

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

The proposed amendment does not alter the design, configuration, or method of operation of the plant. No physical changes are being proposed, nor any changes to the method of operation of the EDGs or supporting systems. The proposed amendment, in effect, allows a small increase in the duration that the EDGs are operated parallel to the grid for test purposes. No new system interactions are created, and the proposed change does not introduce a new failure mode.

Therefore the proposed change does not create the possibility of a new or different kind of accident.

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

The proposed change does not affect the Limiting Conditions for Operation or their Bases that are used to establish any margin of safety. The ability of the EDGs to separate from the offsite power source has been designed and tested per Technical Specification requirements. The proposed change does not involve a change to the plant design or operation and does not affect the availability of any of the required power sources, nor the capability of the EDGs to perform their intended safety function.

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

The foregoing analysis demonstrates that the proposed amendment to HNP TS does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident, and does not involve a significant reduction in a margin of safety.

Based upon the preceding analysis, [Carolina Power & Light Company] CP&L concludes that the proposed amendment does not involve a significant hazards consideration.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 11, 2000.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3.1.F.2.a, ''Primary to Secondary Leakage,'' and 4.13.A.3.f, ''Steam Generator Tube Inservice Surveillance,'' based on the prior replacement of the steam generators (SGs). Specifically, the proposed changes would (1) revise the primary to secondary leakage limits and (2) delete requirements associated with tube sleeve repair, steam generator tube denting, F* repair classification and criteria, and (3) modify the associated TS Bases. In addition, the proposed amendment includes several related administrative changes.

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

Changes to SG Primary to Secondary Leakage Limits

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

The proposed reduction in primary to secondary leakage limit and the elimination of the limit for SGs containing sleeved tubes does not affect accident initiators or precursors. The proposed change establishes a primary to secondary leakage limit that is equivalent to the lesser of the primary to secondary leakage limits currently established for SG with and without SG tube sleeves. Reducing the primary to secondary leakage limit does not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Therefore, the consequences of an accident are not increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not modify any plant equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change establishes one limit for primary to secondary limit that is the same as the most restrictive of the two primary to secondary leakage limits that currently exists. The proposed change does not increase the allowable primary to secondary leakage limit.

Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. The proposed change still requires verification that primary to secondary leakage is within the limit at the existing frequency. Since the primary to secondary leakage limit is not increased, dose rates at the site boundary will not be increased. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

Deletion of Provisions Associated With SG Tube Sleeving Repair Method

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

The proposed deletion of the SG tube sleeving provisions does not affect accident initiators or precursors. The proposed change deletes the TS provisions that are not approved for the replacement SGs. Deletion of an unapproved repair method from the TS does not increase the probability of an accident and the proposed change does not increase primary to secondary leakage limits. Consequently, the consequences of an accident are not significantly increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change deletes the TS provisions that are not approved for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

Deletion of Provisions Associated with Steam Generator F Tube Classification*

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

The proposed deletion of the F* criteria and associated provisions does not affect accident initiators or precursors. The proposed change deletes the TS provisions that are not approved for the replacement SGs. Deletion of an unapproved repair method from the TS does not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change deletes the TS provisions that are not approved for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

Deletion of Provisions Associated With SG Tube Denting Phenomenon

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

The proposed deletion of the requirements and associated provisions regarding SG tube denting does not significantly affect accident initiators or precursors. The proposed change deletes from the TS provisions that are not necessary for the replacement SGs. Deletion of the SG tube denting examination requirements from the TS does not increase the probability of an accident. The proposed change does not increase primary to secondary limits. Therefore, the consequences of an accident are not increased. Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change deletes the TS provisions that are not applicable for the replacement SGs. The proposed change does not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

Related Administrative Changes

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

The proposed administrative changes do not affect accident initiators or precursors. The proposed changes correct the presentation of several TS Basis pages and delete an obsolete scheduler extension footnote. Correcting the page presentation and deleting an obsolete footnote do not increase the probability of an accident. The proposed change does not increase primary to secondary leakage limits. Consequently, the consequences of an accident are not significantly increased.

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

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

The proposed changes do not impact or interface with plant safety related equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed administrative changes do not affect accident initiators or precursors. The proposed change corrects the presentation of several TS Basis pages and deletes an obsolete scheduler extension footnote. The proposed changes do not increase the allowable primary to secondary leakage limit. Since the primary to secondary leakage limit is not increased, the margin of safety will not be reduced. Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) regarding the Limiting Conditions for Operation (LCO) for the auxiliary feedwater system (LCO 3.7.5) to be similar to changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1 (STS), made by the Nuclear Energy Institute Technical Specifications Task Force (TSTF) change number 325, ''Changes To Structure Of [Emergency Core Cooling System] ECCS—Operating LCO.''

Palisades LCO 3.7.5, ''Auxiliary Feedwater System,'' would be changed as follows: (1) An editorial change would be made to Note 2 to put the word ''operable'' in uppercase letters; (2) the second and third parts of the Condition A description, ''AND—At least 100% of the required AFW flow available to each steam generator—AND—At least two AFW pumps OPERABLE,'' would be deleted; (3) the second part of the Condition B description, ''One or more AFT trains inoperable for reasons other than Condition A with at least 100% of the required AFW flow available in MODE 1, 2, or 3,'' would be replaced with two new parts (''Less than 100% of the required AFW flow available to either steam generator—OR—Fewer than two AFW pumps OPERABLE in mode 1, 2, OR 3''); and (4) the wording of

Condition C would be revised to address the condition where insufficient AFW flow is available to achieve a plant shutdown while in any mode within the applicable conditions of LCO 3.7.5. The licensee also forwarded related changes to the TS Bases.

Additional changes requested in the licensee's application dated December 7, 2000, are based upon other TSTFs and are addressed by separate **Federal Register** notices.

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

A. [The proposed changes would not] involve a significant increase in the probability or consequences of an accident previously evaluated.

Changes are proposed to LCO 3.7.5, Auxiliary Feedwater, which emulate changes made to Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Rev.1 (STS) by TSTF 325. The structure of LCO 3.7.5 has been rearranged to maintain Condition A (and, in certain circumstances, Condition B) in effect if failures should occur which reduce available flow to less than 100% of the required flow (that flow assumed in the accident analyses). The resulting requirements are those intended when the LCO was initially constructed and represent the way the LCO Conditions are being applied. Therefore there is no change in intent or application of the LCO. In the case where inoperable AFW train components reduce available flow below that required, and a subsequent partial restoration is made to provide 100% of the required flow, the proposed change makes the literal requirements more conservative because (with the proposed arrangement) the Completion Time for Condition A (and possibly Condition B) would start when the initial inoperability occurred rather than (with literal interpretation of the existing arrangement) when Condition A (or B) was entered after the partial restoration. . . .

As described above, the proposed change corrects the structure of the LCO to assure its correct application. There is no change in intent or in the way the LCO is actually applied. The literal (and unintended) interpretation of the existing LCO structure could, under some circumstances, provide longer than intended Completion Times for restoration of operability. The proposed change only clarifies the requirements of the LCO Required Actions. Since the proposed change affects neither the LCO intent nor its application, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As described above, the proposed change corrects the structure of the LCO to assure its correct application. There is no change in intent or in the way the LCO is actually applied. The proposed changes would not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, no changes to any equipment's function or the method of operating the equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, no new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. [The proposed changes would not] involve a significant reduction in a margin of safety.

As described above, the proposed change corrects the structure of the LCO to assure its correct application. The proposed changes are consistent with the intent of the changes made to the STS by TSTF 325. There is no change in intent or in the way the LCO is actually applied. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1 (STS), made by the Nuclear Energy Institute Technical Specifications Task Force (TSTF) change number 325, ''Changes To Structure Of [Emergency Core Cooling System] ECCS—Operating [Limiting Condition for Operation] LCO.'' Specifically, Palisades LCO 3.5.2, ''ECCS—Operating,'' would be changed as follows: (1) the second part of the Condition B description, ''At least 100% of the required ECCS flow available,'' would be deleted; (2) the wording of Condition C would be revised to limit its application to Conditions A or B; and (3) the wording that would be removed from Condition B would be made into a new condition, Condition D, which would read: ''Less than 100% of the required ECCS flow available.'' Required Action D.1, ''Enter LCO 3.0.3,'' and its completion time, ''Immediately,'' would also be added. The licensee also forwarded related changes to the TS Bases.

Additional changes requested in the licensee's application dated December 7, 2000, are based upon other TSTFs and are addressed by separate **Federal Register** notices.

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 change is proposed which emulates changes made to Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Rev. 1 (STS) by TSTF 325. The structure of LCO 3.5.2, ECCS—Operating, has been rearranged to maintain Condition B in effect if failures should occur which reduce available flow to less than 100% of the required flow (that flow assumed in the accident analyses). The resulting requirements are those intended when the LCO was initially constructed and represent the way the LCO Conditions are being applied. Therefore there is no change in intent or application of the LCO. In the case where inoperable ECCS train components reduce available flow below that required, and a subsequent partial restoration is made to provide 100% of the required flow, the proposed change makes the literal requirements more conservative because (with the proposed arrangement) the Completion Time for Condition B would start when the initial inoperability occurred rather than (with literal interpretation of the existing arrangement) when Condition B was entered after the partial restoration. * * *

A. [The proposed changes would not] involve a significant increase in the probability or consequences of an accident previously evaluated.

As described above, the proposed change corrects the structure of the LCO to assure its correct application. There is no change in intent or in the way the LCO is actually applied. The literal (and unintended) interpretation of the existing LCO structure could, under some circumstances, provide longer than intended Completion Times for restoration of operability. The proposed change only clarifies the requirements of the LCO Required Actions. Since the proposed change affects neither the LCO intent nor its application, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As described above, the proposed change corrects the structure of the LCO to assure its correct application. There is no change in intent or in the way the LCO is actually applied. The proposed changes would not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, and no changes to any equipment's function or the method of operating the equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, no new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. [The proposed changes would not] involve a significant reduction in a margin of safety.

As described above, the proposed change corrects the structure of the LCO to assure its correct application. The proposed change is consistent with the requirements of the STS. There is no change in intent or in the way the LCO is actually applied. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 7, 2000 (this application supercedes an amendment request dated July 28, 2000).

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to allow Type B and C containment leak rate testing to be performed in accordance with 10 CFR part 50, appendix J, option B. Conversion to Option B affects TS 5.5.14 and Surveillance Requirements (SRs) SR 3.6.1.1, SR 3.6.1.3, and SR 3.6.2.1. The proposed amendment also revises the SR 3.6.2.2 frequency for containment air lock door interlock testing from 18 months to 24 months.

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

* * * Four groups of changes have been proposed:

First, changes are proposed to allow Type B and C containment leak rate testing to be performed in accordance with 10 CFR 50, appendix J, Option B.

Second, exceptions are proposed to the Option B testing methodology for containment air lock door seals.

Third, an exception is proposed to the Option B testing frequency for small diameter containment purge valves.

Fourth, the frequency for the containment air lock door interlock testing has been extended from 18 months to 24 months.

The following evaluation supports the finding that operation of the facility in accordance with the proposed changes would not:

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

All four groups of proposed changes deal exclusively with testing of features related to containment isolation. The changes only affect testing frequency and methodology. The proposed testing methodologies are acceptable under the existing Technical Specifications. None of the devices involved are assumed as an initiator of any accident previously evaluated. Therefore, operation of the facility in accordance with the proposed changes would not involve a significant increase in the probability of an accident.

1. The first group of proposed changes is based on the model Technical Specifications approved by the NRC staff in TSTF [Technical Specification Task Force] 52, Rev. 3. Test intervals will be established based on performance history of the components tested. The frequency of testing the containment penetrations and containment isolation valves will be extended in accordance with program requirements and 10 CFR 50, appendix J, Option B, with reference to Regulatory Guide 1.163, and NEI [Nuclear Energy Institute] 94–01, Rev 0. The change in risk resulting from the proposed changes was evaluated by the NRC in the rule making process for implementing the Option B requirements and are characterized in NUREG–1493. For Type B and C tests the NRC concluded that the extension of test intervals as allowed by Option B would lead to only minor increases in potential offsite dose consequences. These increases are offset by the expected decrease in worker dose received during Type A, B, and C testing, and were found to be acceptable. Therefore, operation of the facility in accordance with the first group [of] proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The second group of proposed changes would allow air lock door seal leak rate testing to be performed by a seal contact check (for the Emergency Escape Air Lock) or by pressurizing between the door seals at a pressure [greater than or equal to] 10 psig (for the Personnel Air Lock) following door seal contact adjustments. Both proposed alternative testing methods are allowed by existing Technical Specifications (while testing under Option A) and both will result in a continuation of the currently successful testing practice which has provided a high degree of confidence in door seal performance. Plant operating history has shown that air lock door seals which have been successfully tested in accordance with the proposed methodology have passed subsequent full pressure air lock leakage tests in virtually every case.

Since the proposed methodology has been demonstrated to successfully detect leaking door seals, the continued use of that methodology for testing under the requirements of Option B will not cause an increase in the probability of a leaking air lock door seal going undetected. Also, since there will be no increase in the rate of occurance [sic] of undetected leakage due to the continued utilization of current practices under Option B, operation of the facility in accordance with the second group of proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

3. The third proposed change allows the testing frequency for the Containment 4-inch purge exhaust, 8-inch purge exhaust and 12-inch air room supply valves to be consistent with other 10 CFR 50, appendix J, Option B, Type C test intervals and is supported by Palisades design, historical test results and other required testing. This would allow the test interval to be extended to a maximum of 60 months from the 30 month interval allowed without this exception.

The change in risk resulting from the third proposed change is essentially the same as that evaluated by the NRC in the rule making process for implementing the Option B Type C testing requirements, which are characterized in NUREG–1493. As discussed under change 1, above, the NRC concluded that the extension of test intervals as allowed by Option B for Type C testing would lead to only minor increases in potential offsite dose consequences. These increases were found to be acceptable. The third proposed change applies this longer interval to moderate diameter valves in the containment purge system. That longer interval would apply to these valves, without the proposed exception, if they were installed as containment isolation valves in a different system. Furthermore, the 8-inch and 12-inch valves are effectively leak rate tested on a 184 day frequency as part of their required closure verification. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

4. The fourth proposed change only extends the frequency for containment air lock door interlock testing. The proposed change will not affect any parameters or conditions that contribute to the mitigation of previously evaluated accidents. Therefore, operation of the facility in accordance with the fourth proposed change would not involve a significant increase in the consequences of an accident previously evaluated.

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

All four groups of proposed changes deal exclusively with testing of features related to containment isolation. The changes only affect testing frequency and methodology. The proposed testing methodologies are acceptable under the existing Technical Specifications. The proposed changes would not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, no changes to any equipment's function or the method of operating the equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, they would not cause the containment leak rate testing to become an accident initiator. No new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

c. Involve a significant reduction in the margin of safety.

All four groups of proposed changes deal exclusively with testing of features related to containment isolation. The changes only affect testing frequency and methodology. The proposed testing methodologies are acceptable under the existing Technical Specifications. None of the devices involved are assumed as an initiator of any accident previously evaluated. The proposed changes only affect the methodology and frequency of Type B and C testing. The methods for performing the tests are not changed from those specified in existing Technical Specifications. The proposed performance based approach, provided by using Option B to 10 CFR 50, Appendix J, would continue to ensure that the containment leakage rates would not exceed the maximum allowable leakage rates defined in the Technical Specifications and assumed in the accident analysis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) regarding the Limiting Conditions for Operation (LCO) for the containment cooling systems (LCO 3.6.6), the component cooling water system (LCO 3.7.7), and the service water system (LCO 3.7.8) to be similar to changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1 (STS), made by the Nuclear Energy Institute Technical Specifications Task Force (TSTF) change number 325, ''Changes To Structure Of [Emergency Core Cooling System] ECCS—Operating LCO.''

Palisades LCO 3.6.6, ''Containment Cooling Systems,'' would be changed as follows: (1) The second part of the Condition A description, ''AND—At least 100% of the required post accident containment cooling capability available,'' would be deleted; (2) the wording of Condition B would be revised to limit its application to Condition A; and (3) the wording removed from Condition A would be made into a new condition, Condition C, which would read: ''Less than 100% of the required post-accident containment cooling capability available.'' Required Action C.1, ''Enter LCO 3.0.3,'' and its completion time, ''Immediately,'' would also be added. The licensee also forwarded related changes to the TS Bases.

Palisades LCO 3.7.7, ''Component Cooling Water [CCW] System,'' would be changed as follows: (1) The second part of the Condition A description, ''AND—At least 100% of the required CCW post accident capability available,'' would be deleted; (2) the wording of Condition B would be revised to limit its application to Condition A; and (3) the wording removed from Condition A would be made into a new condition, Condition C, which would read: ''Less than 100% of the required post-accident CCW capability available.'' Required Action C.1, ''Enter LCO 3.0.3,'' and its completion time, ''Immediately,'' would also be added. The licensee also forwarded related changes to the TS Bases.

Palisades LCO 3.7.8, ''Service Water System [SWS],'' would be changed as follows: (1) The second part of the Condition A description, ''AND—At least 100% of the required post accident SWS capability available,'' would be deleted; (2) the wording of Condition B would be revised to limit its application to Condition A; and (3) the wording removed from Condition A would be made into a new condition, Condition C, which would read: ''Less than 100% of the required post-accident SWS capability available.'' Required Action C.1, ''Enter LCO 3.0.3,'' and its completion time, ''Immediately,'' would also be added. The licensee also forwarded related changes to the TS Bases.

Additional changes requested in the licensee's application dated December 7, 2000, are based upon other TSTFs and are addressed by separate **Federal Register** notices.

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:

Changes are proposed for three Palisades LCOs structured like LCO 3.5.2 which emulate changes made to Standard Technical Specifications, Combustion Engineering Plants, NUREG 1432, Rev.1 (STS) by TSTF 325. The structure of LCOs 3.6.6, 3.7.7, and 3.7.8 has been rearranged to maintain Condition A in effect if failures should occur which reduce available flow to less than 100% of the required cooling capability (that assumed in the accident analyses). The resulting requirements are those intended when the LCOs were initially constructed and represent the way the LCO Conditions are being applied. Therefore there is no change in intent or application of the LCOs. In the case where inoperable required components reduce available cooling below that required, and a subsequent partial restoration is made to provide 100% of the required cooling, the proposed change makes the literal requirements more conservative because (with the proposed arrangement) the Completion Time for Condition A would start when the initial inoperability occurred rather than (with literal interpretation of the existing arrangement) when Condition A was entered after the partial restoration. * * *

A. [The proposed changes would not] involve a significant increase in the probability or consequences of an accident previously evaluated.

As described above, the proposed changes correct the structure of the subject LCOs to assure their correct application. There is no change in intent or in the way the LCOs are actually applied. The literal (and unintended) interpretation of the existing LCO structure could, under some circumstances, provide longer than intended Completion Times for restoration of operability. The proposed changes only clarify the requirements of the LCO Required Actions. Since the proposed changes affect neither the LCO intent nor their application, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As described above, the proposed changes correct the structure of LCOs 3.6.6, 3.7.7, and 3.7.8 to assure their correct application. There is no change in intent or in the way the LCOs are actually applied. The proposed changes would not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, and no changes to any equipment's function or the method of operating the equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, no new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. [The proposed changes would not] involve a significant reduction in a margin of safety

As described above, the proposed changes correct the structure of the subject LCOs to assure their correct application. The proposed changes are consistent with the changes made to the STS by TSTF 325. There is no change in intent or in the way the LCOs are actually applied. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1, made by the Nuclear Energy Institute Technical Specifications Task Force (TSTF) change number 258, Revision 4. TSTF 258 addresses changes to various Administrative Controls TSs. The licensee proposes the following four changes to the Palisades TSs:

(1) In section 5.2, ''Organization,'' Palisades TS Section 5.5.2e would be revised by deleting the specific detail of working hour limitations (i.e., administrative procedures are used to control working hours).

(2) Also in Section 5.2, TS Section 5.5.2g would be revised by deleting the title for the ''Shift Technical Advisor'' position and by clarifying the requirements for that position.

(3) In TS Section 5.5.4, ''Radioactive Effluent Controls Program,'' sections 5.5.4b, 5.5.4e, and 5.5.4h would be revised to be consistent with 10 CFR part 20.

(4) TS Section 5.7, ''High Radiation Area,'' would be revised to be consistent with 10 CFR Part 20.1601(c) (i.e., the existing TS would be completely replaced by Insert F from TSTF 258).

Additional changes requested in the licensee's application dated December 7, 2000, are based upon other TSTFs and are addressed by separate **Federal Register** notices.

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

A. [The proposed changes would not] involve a significant increase in the probability or consequences of an accident previously evaluated.

All four proposed changes deal exclusively with Administrative Controls. The changes only affect the details of controls placed on the plant staff and their working conditions. The proposed controls are consistent with the requirements approved for STS. None of the controls involved are assumed to be associated with any initiator of, or any mitigating equipment or mitigation actions for any accident previously evaluated. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

All four proposed changes deal exclusively with Administrative Controls. The changes only affect the details of controls placed on the plant staff and their working conditions. The proposed controls are consistent with the requirements approved for STS. The proposed changes would not result in any physical alterations to the plant configuration, no new equipment is added, no equipment interfaces are modified, no changes to any equipment's function or the method of operating the equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, no new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. [The proposed changes would not] involve a significant reduction in a margin of safety.

All four proposed changes deal exclusively with Administrative Controls. The changes only affect the details of controls placed on the plant staff and their working conditions. The proposed controls are consistent with the requirements approved for STS. None of the controls involved are assumed to be associated with any initiator of, or any mitigating equipment or mitigation actions for any accident previously evaluated. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) in accordance with changes to the ''Standard Technical Specifications, Combustion Engineering Plants,'' NUREG 1432, Revision 1 (STS), made by the Nuclear Energy Institute Technical Specifications Task Force (TSTF) change number 287, ''Allowances For Breach Of The Control Room Envelope,'' Revision 5. Specifically, a note would be added modifying TS section 3.7.10, ''Control Room Ventilation (CRV) Filtration,'' to allow the control room boundary to be opened intermittently under administrative control, and a new condition (Condition B) would be added to the Action table for TS section 3.7.10 to allow 24 hours to restore an inoperable control room boundary. A required Action (B.1) would also be added requiring certain preplanned actions to be initiated immediately upon discovery that the containment envelope is inoperable. The subsequent conditions and required actions would be renumbered accordingly and supporting editorial changes would be made to the descriptions for Conditions B and E (to be renumbered as Conditions C and F). The licensee also forwarded related changes to the Bases for TS section 3.7.10.

Additionally, a correction would be made to the Action table for TS Section 3.7.10 by restoring Required Action D.2 (to be renumbered to E.2), which was inadvertently omitted during the prior issuance of the Palisades Improved TSs by Amendment No. 189.

Additional changes requested in the licensee's application dated December 7, 2000, are based upon other TSTFs and are addressed by separate **Federal Register** notices.

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

A. [The proposed changes would not] involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes deal exclusively with allowances to temporarily deviate from the [Limiting Condition for Operation] LCO 3.7.10 requirement (established by [Surveillance Requirement] SR 3.7.10.4) for the control room boundary to be sufficiently air tight to maintain 0.125 inches of water differential when the ventilation system is in the emergency mode of operation. The proposed controls are consistent with the requirements approved for STS. None of the controls involved are assumed to be associated with any assumed initiator of any accident previously evaluated.

The proposed changes do allow temporary (up to 24 hours) relaxation of controls put in place to protect the operators from accidental releases of particulate radioactive materials. The utilization of this temporary allowance is expected to be infrequent, and the controls required when this allowance is utilized maintain the intended radiological protection for the operators in the control room areas. Since the protection of the operators in the control room areas will be provided by alternate means during the exercising of these allowances, there will be no effect on their perceived abilities to mitigate the consequences of an accident.

Therefore, operation of the facility in accordance with the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes deal exclusively with an allowance to temporarily provide radiological protection within the control room boundary by alternative means. The proposed controls are consistent with the requirements approved for STS. The proposed changes would not result in any physical alterations to the operating plant systems, no new equipment is added, no equipment interfaces are modified, no changes to any equipment's function or the method of operating the power generation or accident mitigating equipment are being made. As the proposed changes would not change the design, configuration or operation of the plant, no new or different kinds of accident modes are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

C. [The proposed changes would not] involve a significant reduction in a margin of safety.

The proposed changes deal exclusively with an allowance to temporarily provide radiological protection within the control room boundary by alternative means. The proposed controls are consistent with the requirements approved for STS. None of the controls involved are assumed to be associated with any initiator of, or any mitigating equipment or mitigation actions for any accident previously evaluated. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 8, 2000.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271), on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated December 8, 2000.

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

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

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

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

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

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

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

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

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

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

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

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

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: October 24, 2000.

Description of amendment request: Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station (GGNS) Operating License be amended to revise the GGNS Technical Specifications (TSs), which govern the lube oil inventories for the Division I, II, and III Emergency Diesel Generators (EDGs). The change would increase the lube oil inventories specified in TS 3.8.3 to ensure continued operation of the EDGs under post-accident conditions, and provide additional margin in lube oil consumption calculations. The TS change would account for potential increases in EDG lube oil consumption rates which exceed the nominal consumption rates originally used to determine EDG lube oil requirements to support seven days of EDG operation at rated load conditions.

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

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

The purpose of the emergency diesel generators is to mitigate the consequences of analyzed accidents. Emergency Diesel Engine inoperability or loss of capability has no effect on the probability of any analyzed accident. The reason for this change is to provide added assurance that the engines perform per the design requirements and therefore the consequences of an accident previously evaluated are not increased.

The purpose of the requested change is to regain margin in the lube oil consumption calculations, such that, if increases in consumption should occur in the future, Technical Specifications requirements will still ensure operability of the Diesel Generators. Design Engineering has basically taken the vendor's specified consumption rate and doubled that value to ensure that the newly calculated inventory limit will bound any potential consumption rate increases.

Current calculations using as found consumption rates have shown that the limiting sump volume is on Division III engines and that there is minimal margin left between the actual volume and the calculated volume needed. Therefore, there is a need for an external dedicated storage skid, which is the only physical change to the plant necessary to support this change request. The current licensing basis recognizes that make-up oil may be required at some point during a design basis event. The current Bases for Technical Specification 3.8.3 LCO provides this recognition.

Given the stated purpose and no need for changes to installed plant structures, (other than addition of a new Div[ision] III lube oil storage skid) systems, or components there will be no significant changes to the operation of the facility. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The purpose of the emergency diesel generators is to mitigate the consequences of analyzed accidents; the engines are not accident initiating. Emergency Diesel Engine inoperability or loss of capability cannot create the possibility of a new or different kind of accident from any accident previously evaluated. The reason for this change is to provide added assurance that the engines perform per the design requirements.

The Diesel Engine Lubricating System (DELS) design and operation is unaffected by his change. Recognizing the need for having a make-up inventory and staging a volume readily accessible to the operator will enhance the operator's ability to maintain DG [Diesel Generator] operable. Design Engineering has performed appropriate fire hazards reviews and seismic II/l reviews to assure compliance with current design requirements.

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

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

The current licensing basis requires that the DELS provide seven days of Diesel Generator operation under specified load conditions. This basis was substantiated via calculation using vendor supplied consumption rates of 1.21 (Div[ision] I and II) and 0.6 (Div[ision] Ill) gallons per hour. The current basis recognizes that make-up oil may be required at some point during a design basis event. To ensure this basis is valid for future operations, Design Engineering has recalculated the required inventories based on a more conservative consumption rate. This change will ensure that sufficient lube oil is readily available to support the extended run times under post accident conditions. Therefore, this change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: December 21, 2000.

Description of amendment request: The amendment request proposes to delete the Steam/Feedwater Flow Mismatch coincident with Low Steam

Generator (SG) Water Level reactor trip from the technical specifications. The Steam/Feedwater Flow Mismatch coincident with Low SG Water Level reactor trip was included in the Unit 1 design in order to meet regulatory requirements regarding potentially adverse control and protection system interactions. The amendment request proposes to take credit for the SG Level Median Selector Switch (MSS) installed in 1997 to meet these requirements. The MSS eliminates the potential for an adverse control and protection system interaction and, therefore, eliminates the design requirement for the Steam/Feedwater Flow Mismatch and Low SG Level reactor trip. Appropriate changes to the Bases are also included in the amendment request.

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

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analys[i]s Report remain as previously analyzed. The proposed change does not introduce a new accident initiator nor does it introduce changes to any existing accident initiators or scenarios described in the Updated Final Safety Analys[i]s Report. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip is not credited for accident mitigation in any accident analyses described in the Updated Final Safety Analys[i]s Report. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip was designed to meet the control and protection systems interaction criteria of the Institute of Electric and Electronic Engineers Standard 279. The Median Selector Switch prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip and satisfies the Institute of Electric and Electronic Engineers Standard 279 requirements. As such, the affected control and protection systems will continue to perform their required functions without adverse interaction and the capability to shut down the reactor when required on Low-Low Steam Generator water level to mitigate an accident previously evaluated is unaffected.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The substitution of the Median Selector Switch for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not introduce any new failure modes to the required protection functions. The Median Selector Switch only interacts with the feedwater control system and the Steam Generator Water Level Low-Low protection function is not affected by this change. Isolation devices in the Median Selector Switch circuitry ensure that the Steam Generator Water Level Low-Low protection function is not affected. The Median Selector Switch is designed to reduce the frequency of system failures through utilization of highly reliable components in a design that relies on a minimum of additional equipment. Components utilized in the Median Selector Switch are of a quality consistent with low failure rates and minimum maintenance requirements, and conform to protection system requirements. Furthermore, the design provides the capability for complete unit testing that provides unambiguous determination of credible system failures. It is through these features that the overall design of the Median Selector Switch minimizes the occurrence of undetected failures that may exist between test intervals. Additionally, the reliability of the Median Selector Switch has been shown by Unit 2 operating experience to be acceptable.

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

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this amendment revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The ability of the Steam Generator Water Level Low-Low reactor trip function credited in the safety analysis to protect against a sudden loss of heat sink event is not affected by the proposed change. Since the Steam Generator Low-Low Level trip provides complete protection for all accident transients that result in low steam generator level, eliminating the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not change any safety analysis conclusion for any analyzed accident described in the Updated Final Safety Analys[i]s Report.

The Median Selector Switch prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and low Steam Generator Water Level reactor trip and satisfies the Institute of Electric and Electronic Engineers Standard 279 requirements. The proposed change will enhance safe operation since the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip function removal decreases the challenges to the plant safety systems, decreases the plant surveillance/maintenance activity, and reduces the plant complexity; all resulting in a reduction in the potential for unnecessary plant transients.

The technical specifications continue to assure the applicable operating parameters and systems are maintained within the design requirements and safety analysis assumptions. Therefore, the elimination of this trip function will not result in a significant reduction in the margin of safety as defined in the Updated Final Safety Analys[i]s Report or technical specifications.

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

Attorney for Licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Marsha Gamberoni.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: January 2, 2001.

Description of amendment requests: The proposed amendment would revise Technical Specifications (TS) 3/4.6.2.2.a for the Unit 1 spray additive tank to require a contained volume between 4000 and 4600 gallons of between 30 and 34 percent by weight sodium hydroxide (NaOH) solution. In addition, the proposed amendment would make four types of format changes to the revised Unit 1 page:

1. Reformat the header to include numbered first and second tier TS section titles and a full-width single line to separate the header section titles from the page text.

2. Reformat the footer to include ''COOK NUCLEAR PLANT—UNIT1'' on the left side of the page, ''Page (page number)'' center page, ''AMENDMENT (past amendment numbers, with strikethrough, and ending with the current amendment number)'' on the right side, and a full-width single line to separate the footer from the page text.

3. Delete the double lines under ''LIMITING CONDITION FOR OPERATION'' and ''SURVEILLANCE REQUIREMENTS.''

4. Fully justify the text and change the font.

Basis for proposed no significant hazards consideration determination:

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

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

Adding a maximum limit for the allowed contained volume and [sodium hydroxide] NaOH concentration for the spray additive tank does not increase the probability of occurrence of any accident. The spray additive system cannot initiate any previously analyzed accident. The proposed changes ensure that the spray additive system and the associated containment spray system can perform the accident mitigation functions required during a [loss-of-coolant accident] LOCA or [main steam line break] MSLB event. This action does not affect the initiating frequency of a LOCA or MSLB event. Therefore, the proposed changes do not increase the probability of an accident previously evaluated.

The accidents previously evaluated in Chapter 14 of the Updated Final Safety Analysis Report that are possibly affected by operation of the spray additive system are a loss-of-coolant accident (LOCA) and a main steam line break (MSLB). These postulated accidents are expected to result in a containment spray signal, which then results in the automatic starting of the containment spray pumps and the opening of the valves associated with the spray additive system. The spray additive system adds NaOH to the containment spray water being supplied from the refueling water storage tank (RWST) to adjust the pH of the containment spray and containment recirculation sump solutions.

Following a LOCA, the containment spray water becomes mixed in the containment recirculation sump with ice melt from the ice condenser, reactor coolant from the reactor coolant system (RCS), water being injected to the RCS from the safety injection accumulators, and water being injected to the RCS from the RWST by the emergency core cooling system. Following a MSLB, the containment spray water becomes mixed in the containment recirculation sump with ice melt from the ice condenser and the secondary coolant released from the ruptured steam line.

The existing minimum and proposed maximum limits for the contained volume and NaOH concentration for the spray additive tank ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine from the containment recirculation sump, and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. An increase in pH value to at least 7.0 in the containment recirculation sump during the recirculation phase following a LOCA is consistent with the iodine retention assumptions of the accident analyses. Therefore, the consequences of a LOCA remain unchanged by the proposed changes. For a MSLB, there is no increase in consequences since the containment spray system and containment recirculation sump are not credited for removal and retention of fission products from the containment atmosphere.

The analyses for determining hydrogen generation following a large break LOCA assume a specific pH time-dependent profile for the containment spray and containment recirculation sump solutions. The existing minimum and proposed maximum limits for the contained volume and NaOH concentration for the spray additive tank do not result in an increase in the previously predicted hydrogen generation rates. Therefore, the current hydrogen generation analyses remain bounding.

For both LOCA and MSLB events, the existing minimum and proposed maximum limits for the contained volume and NaOH concentration for the spray additive tank ensure that the pH of the containment spray solution is within the bounds used in evaluations for environmental qualification of required equipment.

Therefore, the proposed changes cannot increase the probability of occurrence 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?

Adding a maximum allowed contained volume and NaOH concentration for the spray additive tank does not create the possibility of an accident of a new or different type than any previously evaluated. The proposed changes ensure that the spray additive system, and the associated containment spray system, can perform the required accident mitigation functions during a LOCA or MSLB event. There are no other types of accidents that can be postulated that would require the use of the spray additive system or the associated containment spray system for mitigation. The proposed changes do not introduce any new association between the spray additive system and any radioactive system, including the RCS. Therefore, emergency operation of the spray additive system, or postulated failures of the spray additive system, cannot initiate any type of accident.

Therefore, the proposed changes do not increase the possibility of a new or different kind of accident than previously evaluated.

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

The proposed limits on maximum allowed contained volume and NaOH concentration for the spray additive tank ensure that the original margin of safety is maintained by ensuring acceptable pH control following a LOCA or MSLB event. Therefore, the proposed changes ensure that the margin of safety is maintained by limiting the maximum pH of the containment spray and containment recirculation sump solutions following a LOCA or MSLB event.

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

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: October 24, 2000.

Description of amendment requests: The proposed amendments would approve an unreviewed safety question allowing the use of new methodology to calculate the transient response to steam generator tube ruptures.

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

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

The proposed change, to adopt a new analytical method to evaluate the effects of an [Steam Generator Tube Rupture] SGTR, does not affect any accident initiators or precursors. As such, the proposed change does not increase the probability of an accident. The proposed change also does not affect the ability of operators to mitigate the consequences of an accident. The proposed change does not impact the design of the affected plant systems such that previously analyzed systems, structures, and components (SSCs) would now be more likely to fail. The changes will not modify plant systems to reduce their design capability during normal operating and accident conditions. The use of the WCAP–10698–P–A methodology to more accurately calculate the flow from the reactor coolant system (RCS) to the SG secondary side following a postulated SGTR does not affect the probability of any analyzed events. The use of the WCAP–10698–P–A methodology does not affect SGTR initiators or precursors. Therefore, incorporating the new methodology does not affect equipment malfunction probability, nor does it affect or create new accident initiators or precursors. Thus, there will be no reduction in the capability of those SSCs in limiting the consequences of previously evaluated accidents.

Additionally, the present methodology for calculating the radiological consequences of a postulated SGTR is conservative when compared with results from the new methodology. As such, the existing licensing basis radiological consequence calculations will be retained. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the [updated final safety analysis report] UFSAR will not be increased. The use of this WCAP methodology and associated computer code for break flow modeling more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to the analysis of operator actions and the associated timing. Such accurate transient response information enables enhancements to be made to the emergency operating procedures (EOPs).

Therefore, the proposed changes cannot increase the consequences or probability of occurrence of an accident previously evaluated.

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

The proposed change does not impact the design of affected plant systems, involve a physical alteration to the systems, or change to the way in which systems are currently operated, such that previously unanalyzed SGTRs would now occur. The change to incorporate the WCAP–10698–P–A methodology does not introduce any new malfunctions; it calculates more accurately the flow from the RCS to the SG secondary side following a postulated SGTR to determine the time available for operator actions to prevent overfilling the affected SG.

Thus, use of the WCAP–10698–P–A methodology does not affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

Therefore, the proposed changes do not increase the possibility of a new or different kind of accident than previously evaluated.

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

The approval of the license amendment will not result in any modifications to affected plant systems that would reduce their design capabilities during normal operating and accident conditions. By using the WCAP–10698–P–A methodology, a more accurate SGTR response is calculated. The improved understanding of the transient response enables enhancements to the EOPs, which provide further assurance that SSCs required for accident mitigation are protected.

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

In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards.

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

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

NRC Section Chief: Claudia M. Craig.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 18, 2000.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated December 18, 2000.

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

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

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

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

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

Therefore, the elimination of PASS requirements from Technical

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

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

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

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

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

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

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

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

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford

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: July 31, 2000.

Description of amendment request: The proposed license amendment will change the method used to determine the Fuel Centerline Melt Linear Heat Rate Limit (FCMLHRL). The proposed change represents a departure from the use of the fixed value of 21 kilowatts per foot for the FCMLHRL, which is being used in the current operating cycle, to a value that will be calculated on a cycle-by-cycle basis using the Siemens Power Corporation (SPC) U.S. Nuclear Regulatory Commission (NRC) approved methodology. Northeast Nuclear Energy Company (the licensee) has evaluated this proposed method of calculating FCMLHRL utilizing the criteria of 10 CFR 50.59. The licensee has determined that this change involves an unreviewed safety question (USQ). The licensee is requesting approval of the USQ.

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

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

This license amendment request deals with changes in the Millstone Unit No. 2 Final Safety Analysis Report (FSAR) due to changing the method used to determine the FCMLHRL. The proposed change represents a departure from the use of the fixed value of 21 kW/ft for the FCMLHRL, which is being used in the current cycle, to a value that will be calculated on a cycle by cycle basis using the SPC approved methodology. This methodology was reviewed and approved by the Nuclear Regulatory Commission (NRC) and is documented in Siemens Power Corporation (SPC) report XN–NF–82–06(P)(A).[ ] The value of the FCMLHRL is verified for each reload, but does not typically change significantly between cycles. This limit is determined for a standard fuel rod. The current enrichment cutbacks in the gadolinia bearing rods limit their relative power such that the maximum FCMLHRL for a gadolinia bearing fuel rod will be sufficiently below the standard fuel rods to prevent centerline melt. In future applications of this methodology, the peak Linear Heat Rates (LHR) calculated from transient analyses will be compared to the FCMLHRL for the cycle. The Local Power Density (LPD) Limiting Safety System Settings (LSSS) verification analysis for future applications will use the cycle dependent FCMLHRL. Therefore, It can be concluded that these FSAR changes are safe and that the cycle specific calculated FCMLHRL has no impact on plant equipment operation. Further more, the change in the method of determining the FCMLHRL only impacts the analytical determination of failed fuel and has no direct impact on the accident scenario. Accordingly, this change cannot affect the likelihood of these events. Therefore, the proposed changes will not increase the probability of occurrence of accidents previously evaluated.

The change in the method of determining the FCMLHRL will continue to conservatively estimate fuel failures. Since the proposed FSAR changes will have no impact on the analysis of the events, they cannot affect the likelihood or consequences of these events. Therefore, the proposed FSAR changes will not increase the consequences of accidents previously evaluated.

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

The proposed FSAR 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. The FSAR changes do not introduce any new failure modes. Therefore, the changes will not increase the probability of a new or different kind of accident from any accidents previously evaluated.

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

The purpose of the proposed changes is to document a change in the method used to determine FCMLHRL in the Millstone Unit No. 2 FSAR. The change in methodology may result in a FCMLHRL that is higher than the previous limit of 21 kW/ft. Therefore, the proposed changes may lead to a reduction of the margin of safety. However, the proposed changes are safe because SPC has justified, using NRC generically approved methodology, that with a higher value of the FCMLHRL the fuel will not experience centerline melt. In other words, a higher FCMLHRL may allow a higher fuel temperature but will continue to protect fuel against centerline melt. Therefore, it can be concluded that the FSAR changes are safe and do not significantly reduce the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 13, 2000.

Description of amendment request: The proposed amendment would change License Condition 2.C.4 to conform to NRC Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements.'' The proposed amendment would also relocate the Fire Protection Program (FPP) elements from the Technical Specifications (TSs) to the licensee-controlled FPP, in accordance with GL 86–10 and GL 88–12, ''Removal of Fire Protection Requirements from 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. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The requested changes are administrative in nature in that they move fire protection requirements from the TS to the FPP and associated implementing procedures following the guidance of NRC Generic Letter (GL) 86–10 and GL 88–12. The requested changes will not revise the requirements for fire protection equipment operability, testing or inspections.

The proposed changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their radiological consequences are not adversely affected.

Therefore, the probability or consequences of an accident previously evaluated are not affected.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The requested changes are administrative in nature in that they move fire protection requirements from the TS to the FPP and associated implementing procedures following the guidance of GL 86–10 and 88–12. The requested changes will not revise the requirements for fire protection equipment operability, testing or inspections.

The proposed changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes.

Therefore the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The requested changes are administrative in nature in that they move fire protection requirements from the TSs to the FPP and associated implementing procedures following the guidance of GL 86–10 and 88–12. The requested changes will not revise the requirements for fire protection equipment operability, testing or inspections. Future changes to the program will be reviewed in accordance with the fire protection license condition to ensure that the ability to achieve and maintain safe shutdown in the event of a fire are [sic] not adversely affected.

Therefore, a significant reduction in the margin of safety is not involved.

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 13, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.8/4.8 to clarify the air ejector offgas activity sample point and operability requirements.

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

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

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes clarify and more completely specify actions and requirements with respect to main condenser offgas activity. Compliance with applicable regulatory requirements will continue to be maintained. The proposed changes do not alter the conditions or assumptions in any of the previous accident analyses. Since the previous accident analyses remain bounding, the radiological consequences previously evaluated are not adversely affected by the proposed changes.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the method of operation of any plant equipment. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Also, there will be no change in types or increase in the amounts of any effluents released offsite.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed changes do not involve a significant reduction in a margin of safety. The proposed changes clarify and more completely specify actions and requirements with respect to main condenser offgas activity. No changes in radioactivity release limits or dose limits are proposed. The changes in actions to be taken if a limit is not met provide an adequate means of ensuring that the health and safety of the public are protected and that potential dose to the public is below regulatory limits. The proposed changes do not involve any actual change in the methodology used in the control of radioactive effluents. The proposed changes also comply with the guidance contained in the STS [standard technical specifications].

Therefore, a significant reduction in the margin of safety is not involved.

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: December 6, 2000.

Description of amendment requests: The proposed license amendments would revise Section 5.0, ''Administrative Controls,'' of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 Technical Specifications (TS) to change the following management titles.

(1) TS 5.1.1 would be revised to replace the titles ''Vice President, Diablo Canyon Operations and Plant Manager,'' and ''Plant Manager,'' with the generic title ''plant manager.''

(2) TS 5.2.1.a, last sentence, would be revised to state: ''These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR [Final Safety Analysis Report] Update.''

(3) TS 5.2.1.b, would be revised to replace the title ''Plant Manager,'' with the generic title ''plant manager.''

(4) TS 5.2.1.c. would be revised to replace the title ''Senior Vice President and General Manager—Nuclear Power Generation,'' with the generic title ''specified corporate officer.''

(5) TS 5.2.2.d would be revised to replace the title ''Plant Manager,'' with the generic title ''plant manager.''

(6) TS 5.2.2.e. would be revised to replace the title ''Operations Director'' with the generic title ''operations manager.''

(7) TS 5.3.1 would be revised to replace the titles ''Radiation Protection Director'' and ''Operations Director'' with the generic titles ''radiation protection manager'' and ''operations manager,'' respectively.

(8) TS 5.5.1.b (second paragraph ''b'') would be revised to replace the title ''Plant Manager,'' with the generic title ''plant manager.''

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

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

This License Amendment Request (LAR) proposes to revise Technical Specification (TS) 5.0, ''Administrative Controls,'' to replace specific management titles with lower case generic titles consistent with Industry/Technical Specification Task force (TSTF) Standard Technical Specification Change Traveler TSTF–65, Revision 1, approved by the NRC on November 10, 1994.

The proposed changes revise TS 5.0 to change management titles from (a) ''Vice President, Diablo Canyon Operations and Plant Manager'' to ''plant manager,'' (b) ''Senior Vice President and General Manager—Nuclear Power Generation'' to ''specified corporate officer,'' (c) ''Radiation Protection Director'' to ''radiation protection manager,'' and (d) ''Operations Director'' to ''operations manager.''

The proposed changes do not eliminate any of the qualifications, responsibilities or requirements for these positions. Each member of the plant staff assigned to these positions shall continue to meet or exceed the minimum qualifications of ANSI/ANS 3.1–1978, Regulatory Guide 1.8, Revision 2, April 1987 (radiation protection manager), or TS 5.2.2.e (operations manager) as required by TS 5.3.1.

The proposed change to replace the title ''Vice President, Diablo Canyon Operations and Plant Manager'' with the generic title ''plant manager'' reflects PG&E's plan to split the responsibilities of the Vice President, Diablo Canyon Operations and Plant Manager, into two positions: (1) Vice President, Diablo Canyon Operations, and (2) Station Director. The Station Director will report to the Vice President, Diablo Canyon Operations. The Station Director will fulfill the responsibilities of the ''Plant Manager'' as described currently in TS and Final Safety Analysis Report (FSAR) Update and will be responsible for overall safe operation of the plant and will have control over those onsite activities necessary for safe operation and maintenance of the plant. This change results in no change to the responsibilities or qualification requirements for this position as specified in the TS.

The remaining changes are administrative changes only that result in no changes in the responsibilities for the positions.

None of the proposed changes have an impact on plant equipment, or on how plant equipment is operated or maintained, and therefore they have no impact on plant accidents.

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

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

The proposed changes revise TS 5.0 to change management titles from (a) ''Vice President, Diablo Canyon Operations and Plant Manager'' to ''plant manager,'' (b) ''Senior Vice President and General Manager—Nuclear Power Generation'' to ''specified corporate officer,'' (c) ''Radiation Protection Director'' to ''radiation protection manager,'' and (d) ''Operations Director'' to ''operations manager.''

The proposed changes do not eliminate any of the qualifications, responsibilities or requirements for these positions.

None of the proposed changes have an impact on plant equipment, or on how plant equipment is operated or maintained, and therefore they have no impact on initiation of new or different plant accidents.

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

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

The proposed changes revise TS 5.0 to change management titles from (a) ''Vice President, Diablo Canyon Operations and Plant Manager'' to ''plant manager,'' (b) ''Senior Vice President and General Manager—Nuclear Power Generation'' to ''specified corporate officer,'' (c) ''Radiation Protection Director'' to ''radiation protection manager,'' and (d) ''Operations Director'' to ''operations manager.''

The proposed changes do not eliminate any of the qualifications, responsibilities or requirements for these positions.

None of the proposed changes have an impact on plant equipment, or on how plant equipment is operated or maintained, and therefore they have no impact on margin of safety.

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

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 7, 2000.

Description of amendment request: The licensee proposes to revise Technical Specification 3.5.A.1 by adding a note regarding operability of the Low Pressure Coolant Injection system (LPCI) under certain restrictive conditions. The subject change would provide a clarification of system operability that would result in additional flexibility in operations during hot shutdown conditions.

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

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

The LPCI system is not assumed to be the initiator of any previously analyzed event. Its function is in mitigating and thereby limiting consequences of analyzed events. With this proposed change LPCI is still capable of being manually realigned, if needed, to mitigate the consequences of accidents. The allowance provided by this change is only applicable for the reactor in a shutdown condition with reactor pressure less than the RHR [residual heat removal] shutdown cooling permissive setpoint.

Thus, the reactor heat load is much less than assumed for design basis loss of coolant accidents occurring at full power. Furthermore, other emergency core cooling systems are still required to be operable.

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

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

The proposed change does not involve any physical alteration of the plant or introduce new modes of operation. There is no change in plant operation that involves failure modes other than those previously evaluated.

The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change has no impact on any safety analysis assumption. The clarifying Note being added to Technical Specification 3.5.A.1 allows the decay heat removal function to be available without immediate shutdown requirements for inoperable LPCI subsystems being imposed. This is recognition that the amount of time to realign the RHR system from the decay heat removal function has no significant impact on the margin of safety associated with establishing LPCI injection, because the heat loads under these conditions are far less than assumed in the safety analysis.

Placing the reactor in SDC [Shutdown Cooling] during hot shutdown is a normal and preferred method for removing sensible heat from the reactor. In addition, the change does not alter the availability of other safety systems and the ability to meet their safety functions. The additional flexibility, to allow LPCI subsystems to be considered operable during SDC below the RHR shutdown cooling permissive pressure and without entering a shutdown LCO [limiting condition for operation] will not significantly reduce margins of safety since the reactor is in hot shutdown with all control rods inserted, reactor pressure is less than the RHR shutdown cooling permissive pressure, and other ECCS [Emergency Core Cooling Systems] systems should be capable of providing the required cooling, thereby allowing operation of RHR SDC when necessary. Thus, the margins of safety for such situations are maintained.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 19, 2000.

Description of amendment request: The proposed change would revise the reactor vessel pressure/temperature (P/T) limit curves specified in TS 3.6.A.1, "Reactor Coolant Systems—Pressure and Temperature Limitations," as graphically represented in Figure 3.6.1, for reactor heatup, cooldown, and critical operation, as well as for inservice hydrostatic and leak tests.

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

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

The changes to the calculational methodology for the [pressure/temperature] P/T limits based upon [American Society of Mechanical Engineers] ASME [American Society for Mechanical Engineers Boiler and Pressure Vessel Code] Code Cases N–640 and N–588 provide adequate margin in the prevention of a brittle-type fracture of the reactor pressure vessel (RPV). The Code Cases were developed based upon the knowledge gained through years of industry experience. The experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the subject Code Cases in developing the P/T limit curves will continue to provide adequate protection against nonductile-type fractures of the RPV.

The evaluation for revising the P/T limit curves for 4.46×10^8MWH(t) (32 effective full power years) was performed using the approved methodologies of 10 CFR 50, appendix G. The curves generated from these methods ensure the P/T limits will not be exceeded during any phase of reactor operation. The proposed changes will not affect any other system or equipment designed for the prevention or mitigation of previously analyzed events. Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

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

The proposed changes to the reactor pressure vessel P/T limits do not affect the assumed performance of any system, structure, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T curves is unnecessarily conservative. Accordingly, ASME Code Cases N–640 and N–588 take advantage of the acquired knowledge by establishing more enhanced methodologies for the development of P/T curves. Therefore, operational flexibility can be gained without a significant reduction in the margin of safety to RPV brittle fracture.

The revised evaluation of the P/T curves to 4.46×10^8MWH(t) was performed per the guidelines of 10 CFR 50, and thus, the margin of safety is not reduced as the result of 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.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 14, 2000

Description of amendment request: The proposed changes will modify the Technical Specifications Section 3/4.7.7 "Control Room Emergency Habitability Systems" Surveillance Requirements 4.7.7.1.d.1 and 4.7.7.2.a, to revise the differential pressure limit across the control room emergency ventilation system filter assembly and increase the minimum number of compressed air bottles in the control room bottled air pressurization 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:

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

Increasing the minimum required number of air bottles in the control room bottled air pressurization system in order to maintain system capacity does not change the operation of the plant. The control room bottled air pressurization system and the emergency ventilation system will not be operated differently. No new accident initiators are established as a result of the proposed changes. Revising the differential pressure acceptance criteria and including [the] demister filter along with the HEPA filter and charcoal adsorber will provide increased assurance of system readiness. These systems will continue to be operable to limit control room dose to within the analysis of record. Therefore, the probabaility of occurrence or the consequences of an accident previously evaluated is not increased.

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

The proposed changes do not affect the operation of the plant. The control room bottled air pressurization system and control room emergency ventilation system will not be operated differently as a result of the proposed changes. No new accident or event initiators are being created by these changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

[3.] Involve a significant reduction in the margin of safety as defined in the bases [of] any Technical Specifications.

The proposed changes reflect conservative changes in the operating requirements for the control room bottled air pressurization and control room emergency ventilation systems. These changes will further ensure the systems will continue to be operable to mitigate the consequence of an accident for the control room operators. Therefore, the proposed changes do not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: February 28, 2000, as supplemented by letters dated May 12, May 24, June 1, and June 28, 2000.

Brief description of amendment: The amendment revised certain license conditions to reflect the change in ownership interest from PECO to Exelon Generation Company, LLC.

Date of issuance: January 12, 2001.

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

Amendment No.: 228.

Facility Operating License No. DPR–50: Amendment revised the License.

*Date of initial notice in **Federal Register:*** April 10, 2000 (65 FR 19029).

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

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: October 12, 2000.

Brief description of amendment: The amendment changes the name of the facility from WNP–2 to Columbia Generating Station in all applicable locations of the Operating License, appendix A Technical Specifications, and appendix B Environmental Protection Plan. In addition, the proposed action would make editorial changes to TS Figure 4.1–1, ''Site Area Boundary'' modifying or deleting text associated with references to WNP–2.

Date of issuance: January 8, 2001.

Effective date: January 8, 2001, and shall be implemented within 30 days from the date of issuance.

Amendment No: 169.

Facility Operating License No. NPF–21: The amendment revised the Operating License, appendix A Technical Specifications, and appendix B Environmental Protection Plan.

*Date of initial notice in **Federal Register:*** November 29, 2000 (65 FR 71134).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2001.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: November 23, 1999, as supplemented by letters dated February 24 and October 19, 2000.

Brief description of amendment: The amendment incorporated the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the Arkansas Nuclear One, Unit No. 1, Technical Specifications.

Date of issuance: December 28, 2000.

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

Amendment No.: 210.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 8, 2000 (65 FR 12291). The application was renoticed on March 22, 2000 (65 FR 15378).

The October 19, 2000, supplemental letter provided clarifying information and revised Bases pages that was within the scope of the application and did not change the associated no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 10, 1999, as supplemented by letters dated November 4, 1999, and October 12, 2000.

Brief description of amendment: By letter dated June 10, 1999, FirstEnergy submitted its response for Davis-Besse Nuclear Power Station to the actions requested in Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999. By letter dated November 4, 1999, FirstEnergy requested changes to the Technical Specifications (TS) sections 3/4.6.4.4, ''Hydrogen Purge System (HPS),'' 3/4.6.5.1, ''Shield Building Emergency Ventilation System (SBEVS),'' 3/4.7.6.1, ''Control Room Emergency Ventilation System (CREVS),'' and 6.0, ''Administrative Controls,'' for Davis-Besse Nuclear Power Station. FirstEnergy proposes adoption of a Ventilation Filter Testing Program (VFTP) in TS section 6.0—Administrative Control and removal of the specific ventilation filter testing requirements from the plant's Surveillance Requirements of TS sections 3/4.6.4.4, 3/4.6.5.1, and 3/4.7.6.1. By letter dated October 12, 2000, FirstEnergy provided additional information regarding relative humidity in the control room. The proposed changes would revise the TS surveillance testing of the safety related ventilation system charcoal to meet the requested actions of GL 99–02.

Date of issuance: January 11, 2001.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 244.

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

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73091).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 11, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 28, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System,'' to establish actions to be taken for an inoperable control room ventilation system due to a degraded control room boundary (CRB). This revision approves changes that would allow up to 24 hours to restore the CRB to operable status when two control room ventilation system trains are inoperable due to an inoperable CRB in MODES 1, 2, 3, and 4. In addition, a Limiting Condition for Operation note would be added to allow the CRB to be opened intermittently under administrative controls without affecting control room ventilation system operability. Various other editorial changes have been made to reflect the revised TS. The applicable TS Bases have been revised to document the TS changes and to provide supporting information.

Date of issuance: January 2, 2001.

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

Amendment No.: 254.

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

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46010).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 2, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: November 10, 1999, as supplemented October 3, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.7.c, to commit to the American Society for Testing and Materials D3803–1989 test protocol for the ventilation filter testing program. The changes are consistent with Nuclear Regulatory Commission (NRC) Generic Letter 99–02.

Date of issuance: December 27, 2000.

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

Amendment No.: 235.

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

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

The supplemental information in the October 3, 2000, letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 27, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, PSEG Nuclear LLC, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 10, 2000.

Brief description of amendments: The amendments revised the licenses for Peach Bottom Units 2 and 3 to remove Delmarva Power and Light Company as a licensee, in conjunction with the transfer of the minority ownership interests of Delmarva Power and Light Company to the majority owners, PECO Energy Company and PSEG Nuclear LLC.

Date of issuance: December 29, 2000.

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

Amendments Nos.: 238 & 241.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the License.

*Date of initial notice in **Federal Register:*** November 27, 2000 (65 FR 70740).

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

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

Date of application for amendments: December 20, 1999, as supplemented

February 11, February 25, and October 10, 2000.

Brief description of amendments: The amendments revised Facility Operating Licenses DPR–70 and DPR–75 to reflect changes related to the transfer of the license for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, to the extent held by Delmarva Power and Light Company, to PSEG Nuclear Limited Liability Company.

Date of issuance: December 29, 2000.

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

Amendment Nos.: 240 and 221.

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

*Date of initial notice in **Federal Register:*** February 18, 2000 (65 FR 8452). The February 11, February 25, and October 10, 2000, supplements did not expand the scope of the original application with respect to both the proposed transfer action and the proposed amendment action as initially noticed in the **Federal Register**. No hearing requests or comments were received. In addition, the submittal did not affect the applicability of the Commission's generic no significant hazards consideration determination set forth in 10 CFR 2.1315.

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

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 application for amendments: October 13, 1999, as supplemented by letter dated June 1, 2000.

Brief description of amendments: The amendments revised the Technical Specifications to permit relaxation of allowed bypass test times for Limiting Conditions for Operations (LCO) 3.3.1, ''Reactor Trip System Instrumentation'', and LCO 3.3.2, ''Engineered Safety Feature Actuation System Instrumentations''. These changes specifically revise the completion times from 6 hours to 72 hours for inoperable analog instruments, increase bypass times from 6 hours to 12 hours for surveillance testing of analog channels, and increase completion times from 6 hours to 24 hours for an inoperable logic cabinet or master and slave relays.

Date of issuance: December 22, 2000.

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

Amendment Nos.: 116 and 94.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46016).

The supplemental letter dated June 1, 2000, provided clarifying information that did not change the scope of the October 13, 1999, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 17th day of January 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–1987 Filed 1–23–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request; Upon Written Request Copies Available From: Securities and Exchange Commission, Office of Filing and Information Services, Washington, DC 20549

Extension: Rule 17Ad–2(c), (d), and (h), SEC File No. 270–149, OMB Control No. 3235–0130

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget a request for extension of the previously approved collection of information discussed below.

- Rule 17Ad–2(c), (d), and (h) Transfer Agent Turnaround, Processing and Forwarding Requirements.

Rule 17Ad–2(c), (d), and (h), 17 CFR 240.17Ad–2(c), (d), and (h), under the Securities Exchange Act of 1934, enumerate the requirements with which transfer agents must comply to inform the Commission or the appropriate regulator of a transfer agent's failure to meet the minimum performance standards set by the Commission rule by filing a notice.

While it is estimated there are 900 transfer agents, approximately ten notices pursuant to 17Ad–2(c), (d), and (h) are filed annually. The estimated annual cost to respondents is minimal. In view of: (a) the readily available nature of most of the information required to be included in the notice (since that information must be compiled and retained pursuant to other Commission rules); (b) the summary fashion that such information must be presented in the notice (most notices are one page or less in length); and (c) the experience of the staff regarding the notices, the Commission staff estimates that, on average, most notices require approximately one-half hour to prepare. The Commission staff estimates a cost of approximately $30.00 for each half hour spent preparing the notices per year, transfer agents spend an average of five hours per year complying with the rule at a cost of $300.

The retention period for the recordkeeping requirement under Rule 17Ad–2(c), (d), and (h) is not less than two years following the date the notice is submitted. The recordkeeping requirement under this rule is mandatory to assist the Commission in monitoring transfer agents who fail to meet the minimum performance standards set by the Commission rule. This rule does not involve the collection of confidential information. Please note that a transfer agent is not required to file under the rule unless it does not meet the minimum performance standards for turnaround, processing or forwarding items received for transfer during a month. Persons should note that an agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid control number.

Written comments regarding the above information should be directed to the following persons: (i) Desk Officer for the Securities and Exchange Commission, Office of Information and Regulatory Affairs, Office of Management and Budget, Room 10102, New Executive Office Building, Washington, DC 20503; and (ii) Michael E. Bartell, Associate Executive Director, Office of Information Technology, Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549. Comments must be submitted to OMB within 30 days of this notice.

Dated: January 12, 2001.

Jonathan G. Katz,

Secretary.

[FR Doc. 01–2124 Filed 1–23–01; 8:45 am]

BILLING CODE 8010–01–M