significant nonradiological environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission has concluded there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed action, the staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for Point Beach.

Agencies and Persons Consulted

In accordance with its stated policy, on March 11, 1996, the staff consulted with the Wisconsin State official, Ms. Sarah Jenkins, of the Public Service Commission of Wisconsin, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

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

For further details with respect to the proposed action, see the licensee's letter dated October 23, 1995, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Dated at Rockville, Maryland, this 20th day of March 1996.

For the Nuclear Regulatory Commission.

Gail H. Marcus,

Director Project Directorate III–3, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–7408 Filed 3–26–96; 8:45 am]

BILLING CODE 7590–01–P

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 March 4, 1996, through March 15, 1996. The last biweekly notice was published on March 13, 1996.

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

By April 26, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing

Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

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

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

Date of amendments request: February 1, 1996

Description of amendments request: The proposed amendment would (1) revise Technical Specifications (TS) Sections 3/4.1.1.1, 6.9.1.9, and 6.9.1.10 to relocate the shutdown margin (reactor trip breakers open) to the Core Operating Limits Report (COLR); (2) revise TS 3/4.3.2 (Tables 3.3-3 and 3.3-4), to specify an additional restriction for the allowed low pressurizer pressure trip setpoint when reducing reactor coolant system (RCS) pressure in Mode 3; (3) revise TS Section 2.2.1 (Table 2.2-1) to make it consistent with the footnote in TS Tables 3.3-3 and 3.3-4; and (4) revise TS Sections 3/4.5.2 and 3/4.5.3 to specify an additional restriction to require that two emergency core cooling system (ECCS) subsystems be operable in Mode 3 whenever the RCS cold leg temperature is equal to or above 485 degrees F. In addition, the Table of Contents and the Bases would be revised to be consistent with these changes.

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

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed changes to TS Tables 2.2-1, 3.3-3, and 3.3-4 to add additional restrictions to the pressurizer pressure - low trip setpoint requirements are more conservative than the current Technical Specifications and will reflect the updated Mode 3 steam line break safety analyses assumptions. The proposed changes to TS sections 3/4.5.2 and 3/4.5.3 to add additional restrictions to the requirement to have two ECCS Subsystems operable are also more conservative than the current Technical Specifications and will reflect the updated Mode 3 steam line break safety analyses assumptions. Since these changes are more restrictive, they would not contribute to the initiation of any accident, nor would they increase the consequences of an accident, but they would enhance the plant response to a steam line break in Mode 3 to reduce consequences. The proposed changes to relocate the shutdown margin - reactor trip breakers open to the COLR will have no effect on the initiation or consequences of an accident. The shutdown margin-reactor trip breakers open, which would be determined using NRC approved analytical methods, as required by the proposed changes, would ensure that the probability and consequences of an accident would not increase. The changes to the titles of TS 3/4.5.2 and 3/4.5.3, and to the Table of Contents, are editorial and have no effect on the operation of the plant or on any structures, systems or components.

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

The proposed TS changes do not create the possibility of an accident of a new or different kind. The proposed changes to TS Tables 2.2-1, 3.3-3, and 3.3-4, and TS section 3/4.5.2 and 3/4.5.3, to add additional restrictions to the pressurizer pressure - low trip setpoint requirement and add additional restrictions to the requirement to have two ECCS Subsystems operable are more conservative than the current Technical Specifications and will reflect the updated Mode 3 steam line break safety analyses assumptions. Since these changes are more restrictive, and therefore bounded by the current TS, they would not contribute to the initiation of any kind of new or different accident. The proposed changes to relocate the shutdown margin -reactor trip breakers open to the COLR will have no effect on the possibility of a new or different kind of accident. The shutdown margin-reactor trip breakers open, which would be determined using NRC approved analytical methods as required by the proposed changes, would ensure that there would be no possibility of a new or different kind of accident from any accident previously evaluated. The changes to the titles of TS 3/4.5.2 and 3/4.5.3, and to the Table of Contents, are editorial and have no effect on the operation of the plant or on any structures, systems or components.

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

The proposed TS changes do not involve a reduction in any margin of safety. The proposed changes to TS Tables 2.2-1, 3.3-3, and 3.3-4, and TS section 3/4.5.2 and 3/4.5.3, to add additional restrictions to the pressurizer pressure - low trip setpoint requirement and add additional restrictions to the requirement to have two ECCS Subsystems operable are more conservative than the current Technical Specifications and will reflect the updated Mode 3 steam line break safety analyses assumptions. Since these changes are more restrictive, they do not involve a reduction in any margin of safety as currently established by the existing TS. The proposed changes to relocate the shutdown margin - reactor trip breakers open to the COLR will have no effect on any margin of safety. The shutdown margin - reactor trip breakers open would be determined using NRC approved analytical methods as required by the proposed changes, thus ensuring that there would be no reduction in any margin of safety. The changes to the titles of TS 3/4.5.2 and 3/4.5.3, and to the Table of Contents, are editorial and have no effect on the operation of the plant or on any structures, systems or components.

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

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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 Project Director: William H. Bateman

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: February 15, 1996

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7 to add operability requirements for the Keowee Hydro units during periods of commercial power generation. These requirements are based on lake level and power level of the Keowee Hydro units. Also, two surveillance requirements would be added to TS 4.6 to (1) address periodic testing of the circuitry that was added by the modification approved in NRC's SER dated August 15, 1995, and (2) add a load rejection surveillance to ensure that the response of the Keowee Hydro units is bounded by the design criteria used to develop the Keowee operating restrictions.

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

(1) [Does not] involve a significant increase in the probability or consequences of an accident previously evaluated:

Each accident analysis addressed within the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the change proposed within this amendment request. The probability of any Design Basis Accident (DBA) is not significantly increased by this change. In addition, the consequences of the accidents are within the bounds of the FSAR analyses.

The design basis of the auxiliary electrical systems is to supply the required engineered safeguards (ES) loads of one unit and the safe shutdown loads of the other two units. The systems are arranged so that no single failure will jeopardize plant safety. The addition of the operability requirement and surveillances for the Keowee Hydro units will ensure that the electrical systems can meet their design basis.

(2) [Does not] create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

Addition of the operability requirement and surveillances will not create a new or different kind of accident. The addition of the circuitry which is covered by the operability requirement and surveillances has been reviewed and approved by the NRC. Therefore, operation of ONS [Oconee Nuclear Station] in accordance with this Technical Specification amendment will not create any failure modes not bounded by previously evaluated accidents. Consequently, this change will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

(3) [Does not] involve a significant reduction in a margin of safety:

The design basis of auxiliary electrical systems is to supply the required ES loads of one Unit and safe shutdown loads of the other two units. The ability of the Keowee Hydro units to provide emergency power following an accident during a period of Keowee Hydro commercial power generation was reviewed and approved by the NRC in [an] SER dated August 15, 1995. Therefore, there will be no significant reduction in any margin of safety.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: February 20, 1996

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) 3.1.5, 3.1.10, and 4.1. The TS changes would: (1) reduce the frequency for the concentrated boric acid storage tank boron concentration surveillance, (2) delete the chemical and radiochemical surveillance requirements for the reactor coolant for Sr≅189 and Sr≅190, gross beta activity, gross alpha activity, dissolved gas concentration in the reactor coolant, and gross beta activity in the steam generator feedwater, and (3) relocate the surveillance requirements for tritium, chloride, fluoride and oxygen to the Selected Licensee Commitments (SLC) Manual. The proposed changes would also delete some temperature and pressure requirements on control rod operation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The licensee has determined that operation of the facility in accordance with the proposed amendments would not:

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

Each accident analysis addressed within the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the proposed amendment request. The probability of any Design Basis Accident (DBA) is not significantly increased by the proposed amendment due to the fact that the identified cause in the FSAR accidents is not impacted. In addition, the consequences of the accidents are within the bounds of the FSAR analyses since the proposed amendment does not change the accident analysis methods or assumptions described in the FSAR.

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

The proposed amendment revises and eliminates several of the RCS [Reactor Coolant System] chemistry Technical Specification surveillance requirements. The changes in the surveillance requirements do not alter the plant safety features or the method of operation at ONS [Oconee Nuclear Station]. Therefore, operation of ONS in accordance with the proposed Technical Specification will not create any failure modes not bounded by previously evaluated accidents.

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

The proposed amendment does not impact the mitigation of any of the accidents analyzed in the FSAR. Therefore, there is not a significant reduction in the margin of safety associated with the proposed amendment.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: February 22, 1996

Description of amendment request: The licensee has proposed to increase the safety function lift setpoint tolerances for the safety and relief valves that are listed in Surveillance Requirement 3.4.4.1 (Page 3.4-10) of the Technical Specifications TSs) for the Grand Gulf Nuclear Station, Unit 1. The tolerances would be increased from the current plus/minus 1 percent of the safety function (i.e., safety relief valve) lift setpoint to plus/minus 3 percent.

The frequency of verifying these setpoints would not be changed by this amendment request. Also, the other surveillance requirements in the TSs on these valves and the number of these valves required to be operable are not being changed by this 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 (NSHC) in Attachment 2 to its application of February 22, 1996.

In its application, the licensee stated that it has used the NRC staff's safety evaluation report (SER), NEDC 31753-P-A, issued in the NRC letter of March 8, 1993, which evaluated General Electric (GE) topical report NEDC-31753P, ''BWROG [BWR Owners' Group] In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' dated February 1990.

The licensee's NSHC analysis is presented below:

Entergy Operations, Inc. is proposing that the Operating License for Grand Gulf Nuclear Station (GGNS) be amended to increase the tolerance of the safety function lift setpoints [from plus/minus 1%] to plus/minus 3%. The GGNS Inservice Testing (IST) program controls the frequency of safety relief valve (S/RV) testing as required by the GGNS Operating License; therefore, this proposal will also incorporate changes [concerning the setpoint tolerances] to applicable IST procedures. GGNS will incorporate the recommendations of the NEDC-31753-P-A [NRC staff's] SER, by resetting the safety function [S/RV] lift setpoints for all tested valves to within plus/minus 1% of the design lift setpoint and increasing the test sample size by two valves for each valve found outside the plus/minus 3% safety function lift setpoint. S/RV test sample population will be determined based upon the currently licensed ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code.

The commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves no significant hazards if the 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.

Entergy Operations, Inc. has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10CFR50.91(a), Entergy Operations, Inc. is providing the following analysis of the proposed amendment against the three standards in 10CFR50.92(c):

a. No significant increase in the probability or consequences of an accident previously evaluated results from this change.

The GGNS safety design bases for the S/RVs are:

) Prevent overpressurization of the nuclear system that could lead to failure of the reactor coolant pressure boundary,

) Provide automatic depressurization for small breaks in the nuclear system,

) Permit verification of operability,

) Withstand adverse combinations of loadings and forces during abnormal, accident, or special event conditions.

The most limiting vessel overpressurization event is a closure of all main steam isolation valves with a high flux scram. This event was analyzed for GGNS using the minimum number of S/RVs required by the GGNS Operating License. The safety function lift setpoint tolerance used in the analysis bounds the proposed plus/minus 3% setpoint tolerance. The analysis indicates that the S/RVs are capable of maintaining adequate margin below the Operating License Reactor Coolant System Pressure of 1325 psig.

Anticipated operational transients can also challenge the operation of the S/RVs, for instance, Generator Load Reject without Bypass. Analyses have been performed on the limiting events that bound other pressure transient events using safety function limit setpoint tolerances that bound the proposed plus/minus 3% tolerance request. Fuel operating limits are based on the results of these analyses; therefore, adequate fuel thermal margin is maintained.

Plant transients and events that require the use of automatic depressurization and the low-low set feature utilize the relief mode of S/RV operation. This proposed change does not affect the relief mode of S/RV operation.

The verification of valve operability will still be performed in accordance with the GGNS Inservice Testing Program, and S/RV safety mode operability will be verified prior to reinstallation. Analysis of the loads placed on each S/RV sub-system (discharge piping, spargers and associated components) verifies that adequate margin exists to ensure that the overpressurization system can perform its designed function.

The negative tolerance of the safety function lift setpoint remains above the highest setpoint of the S/RV relief mode, and therefore normal vessel pressure. This margin provides reasonable assurance that inadvertent opening of an S/RV will not occur during power operations.

GGNS will replace each S/RV removed for IST program testing with an S/RV that has been reset to within plus/minus 1% of the designed safety function lift setpoint. During each refueling outage, at least six of the installed S/RVs will be tested for safety lift setpoint in accordance with the current IST program plant procedures. This sample population is in agreement with the current ASME Boiler and Pressure Vessel Code requirements for the GGNS IST program, and is more restrictive than the ANSI/ASME OM-1-1981 requirement upon which the setpoint tolerance was based. For S/RV setpoint testing ([the] as-found [setpoint]), additional valves will be tested if the as-found setpoint is outside plus/minus 3% of its designed safety function lift setpoint. Sample expansion will be consistent with the NEDC 31753-P-A SER requirement of two additional valves per valve failure.

The GGNS UFSAR currently requires at least fifty percent of the installed valves to be removed and tested during each refueling outage. GGNS FSAR Questions & Responses ι 211.49 discusses the bases for this requirement. The concern regarded the performance of S/RVs installed in operating plants at the time of GGNS construction and licensing, and that new plants should have significantly better performing S/RVs. The fifty percent requirement provides a very conservative margin of testing to demonstrate that no common cause of S/RV failure occurs within any one operating cycle. The minimum testing of six valves proposed for each outage, with additional testing for each failure from the initial test population, provides reasonable assurance that no common cause failure is occurring without early detection. [The minimum testing of six valves is in agreement with the current ASME Code requirements and is consistent with the current industry practices that was accepted in the NRC staff's safety evaluation report, NEDC 31753-P-A.]

One of the major factors in the requirement of additional testing population beyond ASME Boiler and Pressure Vessel Code is many of the older plants were experiencing failures with multiple stage pilot operated S/RVs. The safety function of this type of S/RV requires operation of a pilot valve that is susceptible to excessive leakage and corrosive bonding to cylinder walls; thereby preventing proper safety function operation. The GGNS Dikkers S/RVs are direct acting, and do not require the operation of a pilot valve for the safety function. The Dikkers S/RV Instruction Manual recommends ''to replace part of the installed valves each maintenance stop (refueling outage)'', and does not prescribe any particular [number of valves to be tested].

Therefore, no significant increase in the probability or consequences of an accident previously evaluated results from this proposed change.

b. This change would not create the possibility of a new or different kind of accident from any previously analyzed.

The plant specific analyses verify that each S/RV will still perform the intended function of preventing overpressurization of the nuclear system. The vessel will have adequate margin below the Operating License Reactor Coolant System Pressure of 1325 psig, and plant system response will not deviate from the expected sequence of events. Each system, structure, and component that communicates with the reactor vessel has been verified to be within its design and operational margin, and no unanticipated plant transients will occur as a result of the safety lift function setpoint tolerance change.

The negative tolerance of the safety function lift setpoint remains above the highest setpoint of the S/RV relief mode, and therefore normal vessel pressure. This margin provides reasonable assurance that inadvertent opening of an S/RV will not occur during power operations.

This proposed change does not add any new systems, structures or supports, nor does it introduce new S/RV operating modes.

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

c. This change would not involve a significant reduction in the margin of safety.

The increase in the S/RV safety function lift tolerance has been analyzed for bounding limiting events and accident conditions. [The safety function lift setpoint tolerance used in the analysis bounds the proposed plus/minus 3% setpoint tolerance.] No condition exists that reduces the margin of safety on the reactor coolant pressure boundary or any system, structure or component that is required to operate during vessel overpressurization events. Fuel operating limits are based on the results of these analyses; therefore, adequate fuel thermal margin is maintained.

[The negative tolerance of the safety function lift setpoint remains above the highest setpoint of the S/RV relief mode, and therefore normal vessel pressure. This margin provides reasonable assurance that inadvertent opening of an S/RV will not occur during power operations.]

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

Based on the above evaluation, Entergy Operations, Inc. has concluded that operation in accordance with the proposed amendment involves no significant hazards considerations.

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

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

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

NRC Project Director: William D. Beckner

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

Date of amendment request: February 22, 1996

Description of amendment request: The amendment proposes to delete a specification which requires a thorough inspection of the Emergency Diesel Generator (EDG) every 24 months during shutdown. In addition this Technical Specification proposes to delete the phrase ''in any thirty day period'' from a specification concerning Allowed Outage time (AOT).

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:

GPU Nuclear has determined that this [technical specification change request] TSCR poses no significant hazard as defined by the NRC in 10 CFR 50.92.

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence of the consequences of an accident.

The proposed activity deletes the requirement to inspect EDGs during shut down from the Technical Specifications. It further modifies the operability of a single EDG for a limited and defined period of time. These changes do not affect the design or performance of the EDGs or their ability to perform their design function. Analysis using PRA techniques indicates the changes do not significantly increase the probability or consequences of an accident.

2. State the basis for the determination that the activity does or does not create a possibility of an accident or malfunction of a different type than any previously identified in the SAR.

The EDGs are not the source of any accident described in the SAR. These changes do not modify the design or performance of the EDGs and do not affect plant functions or actions. Therefore, the proposed change does not create the possibility of an accident or malfunction of a different type than those previously identified.

3. State the basis for the determination that the margin of safety is not reduced. The proposed changes are designed to improve EDG reliability and availability during shutdown periods by providing flexibility in the scheduling and performance of maintenance. The surveillance intervals are unchanged and operability requirements are only modified to an acceptable degree. The proposed activity does not alter the basis of

any technical specification that is related to the establishment or maintenance of a nuclear safety margin. Therefore, the margin of safety is not significantly reduced by this action.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: February 23, 1996

Description of amendment request: The proposed change to the Technical Specifications would allow the implementation of 10 CFR 50, Appendix J, Option B.

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:

GPU Nuclear has determined that this TSCR [technical specification change request] involves no significant hazards considerations as defined by NRC in 10 CFR 50.92.

The major changes from the existing Oyster Creek Technical Specifications requested in accordance with the Option B requirements:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report.

The proposed change implements Option B of 10 CFR 50, Appendix J on performance based containment leakage testing. The proposed change does not involve a change to the plant design or operation. Therefore, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any of the analyzed accidents or malfunctions. The proposed change does request an allowable extension of containment testing. Therefore, a hypothetical leak could remain undetected for a greater period of time. This slight increase in risk has been determined to be insignificant as:

Type A Testing

NUREG 1493 determined that the effect of containment leakage on overall accident risk is small as risk is dominated by accident sequences that result in the failure or bypass of the containment. Industry wide PCILRTs have demonstrated that only a small fraction of the leaks discovered during testing exceeded acceptance criteria, and that the leak rate has been only marginally above the acceptable limit. Only 3% of all leaks can be detected only by PCILRT, therefore, only 3% of the theoretical leaks are affected by the extension to the Type A test interval. Experience at Oyster Creek agrees with the industry wide data in that the majority of the detected leakage from the primary containment is found through Type B and C testing.

NUREG 1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, demonstrates that increasing the Type A leakage test intervals would have a minimal impact on public risk.

Type B and C Testing

Penetrations are designed to ensure reliability of the containment isolation function. Type B penetrations use a double passive seal (e.g. o-ring, gasket) and Type C penetrations use a double isolation valve design to ensure reliability of the isolation function. Because valves perform the isolation function actively, they are more likely to fail on demand (e.g. failure to completely close on demand). To address this failure mode, Type C valves are subjected to increased design constraints and testing to ensure both acceptable leak rates and stroke times. The proposed change does not alter the installation, operation, operating environment, or testing method of these valves. Therefore, the proposed change does not introduce any new component failure modes, nor does it affect the probability of occurrence of any existing evaluated failure mode.

The failure of any single penetration barrier (isolation valve or passive seal) does not cause penetration failure. Therefore, a double failure would have to occur to cause a failure of the penetration and affect containment. Additionally, the proposed change does not change the acceptance criteria for acceptable leakage testing.

The proposed change does not alter plant design or operation, nor does it alter the allowable maximum leakage rate limit. Thus, the proposed change does not affect the probability of occurrence nor the consequences of any evaluated accident or malfunction of equipment important to safety.

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

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change only involves the reduction in Type A, B and C test frequencies, and the Type A test pressure.

Type A Testing

The only changes proposed to the Type A testing are to frequency and test pressure. As the proposed test pressure is grater than the existing test pressure, no new type of accident or malfunction is created, and the increase in pressure provides an additional margin of safety. The increase in pressure provides an additional margin of safety. The increase in surveillance interval cannot introduce any new type of accident or malfunction.

The PCILRT is presently performed at 20 psig. Performance of the PCILRT at P_a (35 PSIG) will provide a more direct leak rate for analysis.P_a is the design pressure of the torus (the drywell design pressure is 44 psig, but the torus is non isolable form the drywell. Therefore, P_a will not create the possibility of the failure of the torus due to overpressurization. No new accident modes can be created by extending the test intervals. No safety related functions or components are altered as a result of this change. Therefore, no new accident or malfunction different form those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

Type B and C Testing

The proposed change only deals with the frequency of performing Type B and C testing. It does not change what components are tested or the method of testing. There is no proposed change to the design or operation of the plant. Therefore, no new accident or malfunction different form those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

3. Operation of the facility in accordance with the proposed amendment would not decrease the margin of safety as defined in the bases of the Technical Specifications.

Type A Testing

Except for the method of defining the test frequency and pressure at which the PCILRT is performed, the methods for performing the actual test are not changed. However, the proposed change can increase the probability that an increase in leakage could go undetected for an extended period of time. NUREG 1493 has determined that under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

Type B and C Testing

The proposed change only affects the frequency of Type B and C testing. The methods for performing the actual test are not changed. The design or operation of Type B and C components are not changed. The proposed change will result in a longer interval between tests of good performing Type B and C components.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a, which is defined by the Oyster Creek Technical Specifications to be 1.0 percent by weight of the containment air at 35 psig per 24 hours. The limitation on containment leakage rate is designed to ensure the total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin of safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed Technical Specification change request.

Therefore, the margin of safety as defined in the bases for the Technical Specification will not be reduced.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment requests: February 22, 1996 (AEP:NRC:0659AA)

Description of amendment requests: The proposed amendments would revise the technical specifications to remove the requirement that the Operations Superintendent must hold or have held a Senior Operator License at Cook Nuclear Plant, or a similar reactor. In addition, a mid-level operations manager will only be required to hold a Senior Operator License if the Operations Superintendent does not hold one.

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:

Per 10 CFR 50.92, this proposed change does not involve a significant hazards consideration because the change does not:

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

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

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

Criterion 1

The amendment request does not involve a significant increase in the probability or consequences of [an] accident previously evaluated because the proposed change to the Technical Specification does not affect the assumptions, parameters, or results of any UFSAR [updated final safety analysis report] accident analysis. The proposed amendment does not modify any existing equipment. It is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed change does not involve physical changes to the plant or changes in plant operating configuration. The proposed change updates the requirements for the Operations Superintendent. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The proposed change updates the requirements for Operations Superintendent. There is no reduction in the margin of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: March 12, 1996 (AEP:NRC:1248)

Description of amendment request: The proposed amendment would remove the technical specifications related to shutdown and control rod position indication while in modes 3, 4, and 5. The change would make the Unit 2 technical specifications consistent with the Unit 1 technical specifications and the Standard 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:

Per 10 CFR 50.92, this proposed change does not involve a significant hazards consideration because the change does not:

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

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

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

Criterion 1

The boron concentration in the reactor coolant system will be high enough to assure adequate SDM in modes 3, 4, and 5. The calculation to obtain the required boron concentration takes into account the position of the rods. Shutdown margin is assumed as an initial condition in the safety analysis. The safety analysis establishes a SDM that ensures specified acceptable fuel design limits are not exceeded. As long as the SDM is satisfied, no change in the probability or consequences of an accident previously evaluated will result from the proposed deletion of the ''position indicator - shutdown'' specification. It is noted that this change is consistent with the new ISTS approved by the NRC as NUREG-1431, Rev. 1.

Criterion 2

The ability to insert the control and shutdown rods provided by the rod control system is not affected by the OPERABILITY status of the ARPI system. As mentioned previously, the reactor coolant system boron concentration will be high enough to assure adequate SDM is maintained. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The margin of safety requirements are not affected by the removal of this T/S. The required SDM which is an initial condition in the safety analysis, is unaffected since the reactor coolant system boron concentration is increased to address the potential ''all rods out'' configuration. Based on these considerations, it is concluded that 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.

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

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

NRC Project Director: John N. Hannon

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: November 29, 1995

Description of amendment request: The proposed amendment would modify the Technical Specifications to remove the requirement for additional pressure relief by a residual heat removal (RHR) spring relief valve during low temperature overpressure protection (LTOP) 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:

The proposed change to delete Technical Specification 3.4.D.3b has been evaluated against the standards of 10 CFR 50.92 and has been determined not to involve a significant hazards consideration. This proposed change does not:

1. Involve a significant increase in the probability or consequence of an accident previously analyzed. The Power Operative Relief Valves (PORVs) remain operable to mitigate any LTOP event. Thus, this change does not result in an increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously evaluated. Removing the RHR spring relief valve as an additional relief requirement does not create the possibility of a new or different kind of accident since the proposal involves neither a hardware modification nor the creation of a unique operating condition.

3. Involve a significant reduction in a margin of safety. Removing the RHR spring relief valve as an additional requirement does not change the results of any of the FSAR Chapter 14 events. The PORVs remain operable to maintain the margin of safety.

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

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011NRC Deputy Director: John Zwolinski

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: November 29, 1995

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.14 to decrease the maximum steam generator (SG) primary-to-secondary leakage rate from 0.15 gpm to 0.10 gpm and would modify TS 4.10 by revising the requirements for unscheduled SG tube inspections that are performed on each SG following a primary-to-secondary tube leak.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. A steam generator leakage assumption greater than the proposed 0.10 gpm/SG limit has been used in the FSAR [Final Safety Analysis Report] Chapter 14 safety analyses. Thus, the FSAR Chapter 14 safety analyses remain bounding. Assuring that an adequate leakage limit exists that initiates corrective actions in a timely manner is important to ensuring a steam generator tube rupture event does not take place. This change modifies the steam generator post-leakage testing requirements to focus inspections on leaking tubes and areas likely to produce similar leakage, in lieu of an expanded test campaign of all three steam generators. Without this change, Technical Specifications require inspection of 3% of the tubes in each steam generator. By inspecting the critical areas of the affected steam generator and possibly expanding inspections to the critical areas of the remaining steam generators, the probability and/or consequences of previously evaluated accidents (e.g., steam generator tube rupture) are not increased.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes will not involve a modification to existing hardware at the plant. The decrease in the maximum allowable steam generator primary leakage rate tends to provide additional time for operator action to take place which, if timely enough, would avoid the consequences of a tube rupture event. The proposed inspection campaign requires inspection of the critical area and may be expanded to the other steam generators to ensure that additional tubes will not fail due to similar causes. This modified inspection campaign does not introduce the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety. The FSAR Chapter 14 safety analyses assume a higher steam generator leakage rate and therefore remain conservative. The proposed reduction in the allowable leakage provides a greater margin of safety since it is more conservative than the present value. This change modifies inspection requirements of Technical Specifications and does not impact the plant design or equipment. The modified inspection requirements following a plant shutdown due to tube leakage concentrate steam generator tube inspections in those areas believed to be most susceptible to flaws. For these reasons, we believe the proposed changes increase the margin of safety by inspecting the critical areas of the steam generator(s) in lieu of additional random inspections.

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

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011NRC Deputy Director: John Zwolinski

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

Date of amendment request: November 8, 1995

Description of amendment request: The amendment request would revise the Technical Specifications (TS) for the jet pumps to be consistent with the limiting conditions for operation and surveillance requirements in the Standard Technical Specifications for General Electric Plants (NUREG-1433).

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

...The proposed change does not involve an [significant hazards consideration] SHC because the change would not:

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

The new LCO [Limiting Condition for Operation] does not diminish the existing requirement that all jet pumps must be operable, nor does it affect the time available to achieve cold shutdown should a pump become inoperable. The new LCO does eliminate the ability to continue to operate with the indication (but not the function) of a single jet pump inoperable. This does not increase the possibility of an unnecessary plant shutdown due to inoperable instrumentation since sufficient flexibility exists in the surveillance requirement so that operability of the jet pumps can be verified. This change eliminates the LCO that allowed continued operation with conditions that could potentially mask an inoperable pump. The new LCO is more limiting in ensuring that the plant is operated in a condition for which accidents were analyzed.

The new surveillance requirement provides a more accurate method of ensuring the jet pumps remain operable. The new surveillance criteria are more sensitive to jet pump failures and the degradation of the jet pumps prior to failure.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The new LCO and surveillance does not change the manner in which the plant is operated, nor does it reduce the operability requirements of any jet pump, Therefore, no new or different kind of accident can be created by the new specification. The surveillances that will be performed do not require any new hardware or plant evolutions. Therefore, the proposed change to the LCO and surveillance cannot create the possibility of a new or different kind of accident.

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

The margin of safety that currently exists is not diminished by this change. The requirement to place the reactor in cold shutdown within 24 hours should a jet pump become inoperable is maintained. The LCO which allowed continued operation with indication for one pump inoperable has been eliminated.

The new surveillance requirement continues to demonstrate the operability of the jet pumps and during operation, continues to be performed at the same interval as in the current technical specifications. The note (which allows the surveillance to be deferred until four hours after the associated recirculation loop is in operation and 24 hours after exceeding 25% of rated thermal power) does not significantly affect the margin of safety. The time that the unit would be operating in these conditions would be small, and the stress placed on the pump at less than 25% power is lower.

Based on the above, 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.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: November 3, 1995

Description of amendment request: The proposed amendment will extend the allowed outage time from 48 hours to 7 days for an emergency core cooling system train that is declared inoperable as a result of an inoperable low pressure safety injection subsystem.

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:

Pursuant to 10CFR50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed change to extend the allowed outage time (AOT) for an inoperable low pressure safety injection (LPSI) subsystem from the existing limit of 48 hours to 7 days. In addition, the change to modify the completion time for the Action Statement and the criteria for the Surveillance Requirements were also reviewed. NNECO concludes that these changes do not involve a significant hazards consideration (SHC) since the proposed change satisfies the criteria in 10CFR50.92(c). That is, the proposed change does not:

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

The proposed amendments for Millstone Unit No. 2 will extend the action completion AOT for a single inoperable LPSI train from 48 hours to 7 days. A LPSI subsystem is designed as a part of each emergency core cooling system (ECCS) train to supplement safety injection tank inventory during the early stages of mitigating a design basis accident (DBA). As such, components of the LPSI subsystem are not accident initiators, and an extended AOT to restore operability of an inoperable LPSI subsystem would not increase the probability of occurrence of accidents previously analyzed.

The safety analyses for Millstone Unit No. 2 demonstrates that ECCS performance acceptance criteria are satisfied with only one of the two redundant ECCS trains operating during the postulated DBA. The proposed technical specification revisions involve the AOT for a single inoperable LPSI subsystem, and do not change the conditions assumed for the minimum amount of operating equipment needed for accident mitigation. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

In addition, CE NPSD-995 recognizes that when an ECCS train is inoperable due to a LPSI subsystem being unavailable, due either to being declared inoperable (by failing a surveillance requirement) or is intentionally taken out-of-service (for corrective or preventive maintenance), the core damage frequency (CDF) during power operation increases. The results of the PRA presented in CE NPSD-995 show that the proposed increase in the ECCS AOT (due to LPSI unavailability) from 48 hours to 7 days does not cause a significant increase in the overall CDF of Millstone Unit No. 2.

The analyses indicate that continued plant operation with a single LPSI subsystem out-of-service may result in a small increase in ''at power risk;'' however, that risk increase will be negligibly small and controlled effectively via the Maintenance Rule and the risk monitor program that minimizes the outage time and prevents entering into an unacceptable risk configuration. In addition, the proposed AOT extension for the LPSI subsystem is evaluated as having negligible impact on the large early radiological release probability for Combustion Engineering pressurized water reactors in the event of a design basis accident.

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

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

The proposed amendment will not change the physical plant or the modes of plant operation defined in the technical specifications. The changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. Therefore, operation of Millstone Unit No. 2 in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety associated with the ECCS train is established by acceptance criteria for system performance defined in 10CFR50.46. The proposed amendment will not change this acceptance criteria nor the operability requirements for equipment that is used to achieve such performance as demonstrated in the Millstone Unit No. 2 safety analyses. Moreover, an integrated assessment of the risk impact of extending the AOT for a single inoperable LPSI train has concluded that the risk contribution is small. Therefore, operation of Millstone Unit No. 2 in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: September 12, 1995

Description of amendment request: The amendment would revise and reformat Technical Specification (TS) 6.3.1 to add the requirement that the Assistant Operations Manager shall hold a senior reactor operator (SRO) license if the Operations Manager does not hold an SRO license for Millstone Unit 3. Also the footnote would be deleted from TS 6.3.1 that previously granted a one-time three year exception to the qualification requirements for the Operations Manager and an exception for the Assistant Operations Manager to hold a license instead of the Operations 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:

...The proposed change does not involve an [significant hazards consideration] SHC because the change would not:

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

The proposed change affects an administrative control, which was based on the guidance of ANSI N18.1-1971. ANSI N18.1-1971 recommended that the Operations Manager hold an SRO license. The current guidance in Section 4.2.2 of ANSI/ANS 3.1-1987 recommends, as one option, that the Operations Manager have held a license for a similar unit and the Operations Middle Manager hold an SRO license. While the Operations Middle Manager position does not exist at Millstone Unit No. 3, [Northeast Nuclear Energy Company] NNECO has created the position of Assistant Operations Manager. The individual in this position would meet the requirements for, and would have responsibilities as recommended in, ANSI/ANS 3.1-1987 for the Operations Middle Manager position.

Therefore, the proposed change requests an exception to ANSI N18.1-1971 to allow use of ANSI/ANS 3.1-1987 in a limited circumstance. Specifically, the proposed revision to Technical Specification 6.3.1 would require the Operations Manager to either hold an SRO license at Millstone Unit No. 3 or have held an SRO at a [pressurized water reactor] PWR.

If the Operations Manager does not hold an SRO license at Millstone Unit No. 3, the specification will require the Assistant Operations Manager to hold, and continue to hold, an SRO license. The proposed change includes the requirement for the Operations Manager to have held a license for a similar unit (a PWR) in accordance with Section 4.2.2 of ANSI/ANS 3.1-1987. For those areas of knowledge that require an SRO license, the Assistant Operations Manager will provide the technical guidance normally provided by the Operations Manager.

The proposed change does not alter the design of any system, structure, or component, nor does it change the way plant systems are operated. It does not reduce the knowledge, qualifications, or skills of licensed operators, and does not affect the way the Operations Department is managed by the Operations Manager. The Operations Manager will continue to maintain the effective performance of his personnel and ensure the plant is operated safely and in accordance with the requirements of the operating license. Additionally, the Control Room Operators will continue to be supervised by the licensed Shift Supervisors.

The proposed change does not detract from the Operations Manager's ability to perform his primary responsibilities. In this case, by having previously held an SRO license, the Operations Manager has achieved the necessary training, skills, and experience to fully understand the operation of plant equipment and the watch requirements for operators. In summary, the proposed change does not affect the ability of the Operations Manager to provide the plant oversight required of his position. Thus, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to Technical Specification 6.3.1 does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. It does not affect the performance of NRC licensed operators. Operation of the plant in conformance with technical specifications and other license requirements will continue to be supervised by personnel who hold an NRC SRO license. The proposed change to Technical Specification 6.3.1 ensures that the Operations Manager will be a knowledgeable and qualified individual to have held an SRO license at a PWR. Based on the above, the proposed change does 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.

The proposed change involves an administrative control that is not related to the margin of safety. The proposed change does not reduce the level of knowledge or experience required of an individual who fills the Operations Manager position, nor does it affect the conservative manner in which the plant is operated. The Control Room Operators will continue to be supervised by personnel who hold an SRO license. Thus, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: November 21, 1995

Description of amendment request: The licensee proposes to change Technical Specification Section 1.33 and Bases Sections 3/4.3.3.9 and 3/4.3.3.10, and 3/4.11.2.1. The changes clarify the definition of source check to include a source check from a light emitting diode (LED), as well as from ionizing radiation.

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:

... NNECO concludes that these changes do not involve a significant hazards consideration since the proposed changes satisfy the criteria in 10CFR50.92(c). That is, the proposed changes do not:

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

The proposed changes to the definition of source check clarifies the source check for the liquid and gaseous effluent radiation monitors. These monitors do not provide a safety function and only serve to provide radiological information to plant operators, therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to the definition of source check have no effect on the ability of the monitors to perform their designed function. The clarification to the surveillance do not involve any physical modifications to any equipment, structures, or components. The monitors already have the internal LEDs which were originally used to perform the source check. The proposed changes have no impact on design basis accidents, and the changes will not modify plant response or create a new or unanalyzed event.

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

The proposed changes to the definition of source check do not have any impact on the protective boundaries and, therefore, have no impact on the safety limits for these boundaries. The instrumentation associated with these changes do not provide a safety function and only serve to provide radiological information to plant operators. The instrumentation has no affect on the operation of any safety-related equipment. As such, these changes have no impact on the margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

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

NRC Project Director: Phillip F. McKee

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: February 15, 1996

Description of amendment request: The amendment changes the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, by creating Technical Specification Section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' which refers to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

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

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

The adoption of 10 CFR 50, Appendix J Option B will not involve a significant increase in the probability or consequences of any accident previously evaluated. The proposed changes to the TS [Technical Specifications] reflect the use of the performance-based containment leakage-testing program. The USNRC has approved the use of a performance-based option for containment leakage testing programs when it amended 10 CFR 50, Appendix J (60 FR 49495). For adoption of the revised regulation, licensees are required to incorporate into their TS, by general reference, the USNRC regulatory guide or other plant-specific implementing document used to develop their performance-based leakage testing program. A new Administrative Control subsection (5.5.12, ''Primary Containment Leakage Rate Testing Program'') has been added that requires the establishment and maintenance of a Primary Containment Leakage Rate Testing Program. The TS will still require the performance of a periodic general visual inspection of the containment to ensure early detection of any structural deterioration of the containment that may occur.

As concluded in NUREG-1493, given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by ILRT [Integrated Leak Rate Test] testing, increasing the interval between ILRTs is possible with minimal impact on public risk. Additionally, performance-based alternatives to current LLRT [Local Leak Rate Test] requirements are feasible without significant risk impacts. Additionally, these changes will not alter any safety limits which ensure the integrity of fuel barriers, and will not result in a significant increase to onsite or offsite dose.

No physical changes are being made to the plant, nor are there any changes being made in the operation of the plant as a result of these changes which could involve a significant increase in the probability or consequences of any accident previously evaluated. Additionally, these changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients.

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

The adoption of 10 CFR 50, Appendix J Option B will not create the possibility of a new or different type of accident from any previously evaluated. These changes to the PBAPS, Units 2 and 3 TS will not involve any changes to plant systems, structures or components (SCCs) which could act as new accident initiators. These changes will not impact the manner in which SSCs are tested such that a new or different type of accident from any previously evaluated could be created.

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

No margins of safety are reduced as a result of the proposed adoption of 10 CFR 50, Appendix J Option B. As stated previously, the USNRC has approved the use of this performance-based option for containment leakage testing programs when it amended 10 CFR 50, Appendix J (60 FR 49495). These changes will not impact core limits or any other parameters that are used in the mitigation of a UFSAR [Updated Final Safety Analysis Report] design-basis accident or transient. Additionally, these changes do not introduce any hardware changes, and will not alter the intended operation of plant structures, systems or components utilized in the mitigation of UFSAR design-basis accidents or transients. These changes will not introduce any new failure modes of plant equipment not previously evaluated.

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

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

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

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: January 26, 1996

Description of amendment request: The proposed amendment removes three pressure relief valves from Technical Specification Table 3.6.3-1, ''Primary Containment Isolation Valves,'' since these valves are no longer needed to support the steam condensing mode of the residual heat removal (RHR) system and are being removed from the plant.

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.

With the prior deletion of the steam condensing mode of RHR and the isolation of the high and low pressure interfaces, the three pressure relief valves that are being removed from the plant have no active function. Their passive function of maintaining system or containment integrity will be fulfilled by blind flanges. Also, the RHR and RCIC [reactor core isolation cooling] piping are provided with overpressure protection from other pressure relief valves. Therefore, the removal of these pressure relief valves does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The pressure relief valves that are being removed had two primary functions. First, they provided overpressure protection for the RHR and RCIC piping during the steam condensing mode of RHR. Since the steam condensing mode has been deleted from the plant, these valves no longer have that function. Also, overpressure protection of the RHR and RCIC piping is provided by other existing pressure relief valves. Second, these valves maintained system or containment integrity. When the pressure relief valves are removed from the plant, they will be replaced with blind flanges or equivalent that will maintain system or containment integrity. Therefore, the removal of the three pressure relief valves does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since the steam condensing mode of RHR has been eliminated, the three pressure relief valves have no active function. Their passive function of maintaining system or containment integrity will be fulfilled by blind flanges or equivalent. Also, overpressure protection of RHR and RCIC piping is provided by other existing pressure relief valves. Therefore, the removal of the three pressure relief valves does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

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

NRC Project Director: John F. Stolz

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

Date of amendment request: January 25, 1996

Description of amendment request: The amendment proposes to revise the allowed out-of-service times for single inoperable Emergency Diesel Generators (EDGs) to accommodate on-line maintenance of the EDGs. In addition, two line item changes are proposed: (1) to improve safety by reducing EDG testing at power; and (2) to revise the ac power requirements during cold shutdown or refueling modes to make the James A. FitzPatrick (JAF) Technical Specifications consistent with the Standard 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:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

a. EMERGENCY DIESEL GENERATOR LCO [Limiting Conditions for Operation] AT POWER

The proposed changes to the Technical Specifications will allow longer Allowed Out of Service Times [AOTs] to perform necessary repair and maintenance on individual Emergency Diesel Generators while at power. This extended AOT will enhance scheduling of preventive maintenance of individual EDGs without significantly increasing the probability or consequences of an accident previously evaluated. The risk evaluations contained in the JAF quantitative analyses of the EDGs determined that the probability of an accident by increasing the AOT for an individual EDG from 7 days to 14 days is non-risk-significant. The primary reason for this low relative risk is due to the designed redundancy and capability to respond to an accident when a single diesel generator is out of service. LOCA [loss-of-coolant accident] Analyses that assume the worst case line break while an EDG is out of service indicate the plant can be safely shut down with the remaining EDGs. Even if another EDG should fail during the AOT, at least one Core Spray and one Residual Heat Removal (RHR) Low Pressure Coolant Injection pump can provide the required flow to bring the plant to safe shut down. Furthermore, long term suppression pool and reactor shutdown cooling is provided by any one of the three remaining RHR pumps for a single EDG out of service or by two remaining RHR pumps assuming an additional EDG failure during the AOT.

Increasing the EDG AOT does not involve physical alteration of any plant equipment and does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the increased AOT for a single EDG.

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

b. EMERGENCY DIESEL GENERATOR LCO DURING PLANT SHUTDOWN

Changing the number of EDGs required during plant shutdown does not involve physical alteration of any plant equipment and does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the change in the LCO during shutdown.

c. EMERGENCY DIESEL GENERATOR SURVEILLANCE AT POWER OPERATION

The proposed change to the Technical Specification will reduce the required number of tests to be performed when an EDG or EDG System is inoperable. This proposed change to TS requirements addresses the concern of excessive testing that could result in EDG wear which is counter-productive to safety in terms of equipment degradation and availability. This change is consistent with Generic Letter 93-05 guidance for implementing such recommendations. The proposed Technical Specifications will not result in a change to the design or operation of the facility, therefore, this change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

a. EMERGENCY DIESEL GENERATOR LCO AT POWER

Extending the AOT for an individual EDG does not necessitate physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for JAF plant.

b. EMERGENCY DIESEL GENERATOR LCO DURING PLANT SHUTDOWN

Changing the number of EDGs required during shutdown does not necessitate physical alteration of the plant or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for JAF plant.

c. EMERGENCY DIESEL GENERATOR SURVEILLANCE AT POWER OPERATION

The proposed change does not change design, operation or the testing process. The nature of this change precludes the possibility of a new or different kind of accident. The proposed change to complete the required action does not involve any hardware changes, nor changes to the operation of the equipment nor does it change the ability of the equipment to perform its intended function. Performing the testing on an extended time cannot initiate any type of accident.

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

a. EMERGENCY DIESEL GENERATOR LCO AT POWER

As discussed above, the JAF quantitative evaluation determined that the change in risk associated with extending the AOT for a single EDG is non-risk-significant. In addition, the design provides adequate redundancy for safe shut down during the AOT for a single EDG out of service. This is supported by the LOCA analyses including analyses for long term suppression pool and reactor shutdown cooling.

b. EMERGENCY DIESEL GENERATOR LCO DURING PLANT SHUTDOWN

The margin of safety is not affected by changing the number of EDGs required during shutdown. One offsite power source or one EDG ensure the availability of the required power to recover from postulated accident events during shutdown. When the required number of operable systems is not met, all work that could potentially initiate a postulated accident event during shutdown is suspended.

c. EMERGENCY DIESEL GENERATOR SURVEILLANCE AT POWER OPERATION

The proposed change to Technical Specifications reduces testing at reactor power. The overall effect is a net gain in plant safety by avoiding the potential for unnecessary wear that could degrade the EDGs at power. Implementation of these changes is consistent with the guidance provided by the NRC in Generic Letter 93-05. The proposed change to the EDG testing requirements does not reduce the ability of the equipment to perform its intended safety function.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

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

Date of amendment request: January 30, 1996

Description of amendment request: The proposed Technical Specifications change will delete the requirement that oxygen concentrations for both normal and transient conditions not exceed saturation when the reactor coolant is below 250 degrees F. The Technical Specifications change will also eliminate the surveillance requirement for reactor coolant chemistry sampling of chloride, fluoride, and oxygen concentration during maintenance activities when fuel is removed from the reactor vessel and the Reactor Coolant System (RCS) is drained below the reactor vessel flange regardless of whether the upper internal and/or vessel heat are in place or not. Administrative result of the changes being made, capitalize Technical Specifications defined terms to maintain consistency within the Technical Specifications, and the word ''degrees'' is spelled-out when referring to the Fahrenheit temperature, rather than using the symbol.

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:

Specifically, operation of Surry Power Station in accordance with the proposed changes will not:

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

Since the RCS and the RHR [Residual Heat Removal] System are drained when the RCS inventory is reduced below the reactor vessel flange for maintenance or refueling activities, the concentrations of chlorides and fluorides will not change. During these maintenance or refueling activities, only controlled makeup to the RCS is planned, and any planned or unplanned makeup to the RCS would be detected by available level indication. Sampling for chloride and fluoride concentrations in the RCS will be performed prior to draining the system. Sampling of the reactor coolant for chloride and fluoride concentrations will resume when the RCS is filled. The chloride and fluoride concentrations will be known and will be maintained consistent with the Technical Specification Limiting Condition for Operation and Action Statements. Also, when the RCS inventory is drained below the reactor vessel flange, the RCS is vented and open to the containment building atmosphere with the reactor coolant liquid considered oxygen saturated. Technical Specification 3.1.F.4 allows normal and off-normal ''saturated'' oxygen concentrations when reactor coolant temperature is below 250 degrees F. Consequently, sampling the reactor coolant for oxygen concentration under these conditions is not required and the Technical Specification Table 4.1-2B specified sampling frequency of five (5) times per week is not necessary since the oxygen concentration continues to remain in compliance with the Technical Specification limit, measures are available and action can be taken to correct the condition prior to any deleterious effect.

Surry Technical Specifications 3.1.F.1 prohibits reactor coolant temperature from exceeding 250 degrees F unless chloride, fluoride, and oxygen concentrations are within specified limits. Therefore a significant increase in the probability or consequences of an accident previously evaluated does not exist.

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

The materials that are exposed to reactor coolant are corrosion resistant. They were chosen for specific applications within the system and for their compatibility with the reactor coolant. The chemical composition of the reactor coolant will be maintained within the specifications given within Technical Specification 3.1.F, Updated Final Safety Analysis Report Table 4.2-2, and Technical Specification Table 4.1-2B. Because of the time dependent nature of any adverse affects from chloride, fluoride, and oxygen concentrations in excess of the Technical Specifications limits, measures are available and can be taken to correct the condition while the reactor is in a safe shutdown condition, prior to any deleterious effect. No hardware modifications are involved. System configuration and plant operations are not being changed. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

(3) Involve a significant reduction in the margin of safety.

This change does not involve a significant reduction in the margin of safety since the chloride and fluoride concentrations are maintained within their specified values prior to RCS drain down and following refill. The time period during which the RCS inventory is reduced below the reactor vessel flange and fuel is removed from the vessel, is short and insignificant in terms of the parameters necessary to initiate a corrosion concern. Existing Technical Specifications Action Statements and Allowed Technical Specification values for normal and off-normal concentrations of chlorides and fluorides are not being changed. No hardware modifications are involved. System configuration and plant operations are not being changed. Surry Technical Specification 3.1.F.1 remains unaffected by this change and continues to prohibit reactor coolant temperature from exceeding 250 degrees F unless chloride, fluoride, and oxygen concentrations are within specified limits.

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

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219

NRC Project Director: Eugene V. Imbro

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

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

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket No. 50-498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: February 29, 1996

Description of amendment request: The proposed amendment would include the addition of Technical Specification 3.10.8 which would allow a one-time only extension of the standby diesel generator (SDG) allowed outage time for a cumulative 21 days on ''A'' train SDG. In addition, it would also allow a one-time only extension of the allowed outage time on ''A'' train essential cooling water loop for a cumulative 7 days. This one-time only change would become effective on April 10, 1996, and expire on May 15, 1996.Date of individual notice in the Federal Register: March 8, 1996 (61 FR 9502)

Expiration date of individual notice: April 8, 1996

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 1, 1995, as supplemented by letters dated June 22, August 28, November 22, and December 19, 1995, and January 4, January 8 (two letters), and January 23, 1996

Description of amendment request: The proposed amendment would provide a special test exception that would allow an extension of the standby diesel generator (SDG) allowed outage time for a cumulative 21 days on each SDG once per fuel cycle, and it would also allow an extension of the essential cooling water (ECW) loop allowed outage time for a cumulative 7 days on each ECW loop once per fuel cycle. These extended allowed outage times will be used to perform required inspections and maintenance on the SDGs and the ECW system during power operation.

Date of individual notice in the Federal Register: February 8, 1996 (61 FR 4805)

Expiration date of individual notice: March 11, 1996

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

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: March 1, 1996 (supersedes December 11, 1995, application)

Description of amendment request: The proposed amendment would revise Technical Specification Section 4.7, ''Surveillance Requirements for Primary Containment Automatic Isolation Valves.'' Specifically, the proposed amendment would revise the replacement frequency of the seat seals for the drywell and suppression chamber purge and vent valves from every 5 years to every six operating cycles.

Date of individual notice in the Federal Register: March 8, 1996 (61 FR 9504)

Expiration date of individual notice: April 8, 1996

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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 application for amendments: November 7, 1995, as supplemented by letter dated January 17, 1996.

Brief description of amendments: These amendments adopt the improved Standard Technical Specifications (NUREG-1432) format and content of Section 5.0, ''Design Features,'' as modified by approved changes to the improved Standard Technical Specifications.

Date of issuance: March 6, 1996

Effective date: March 6, 1996, to be implemented within 45 days of the date of issuance.

Amendment Nos.: Unit 1 - Amendment No. 104; Unit 2 - Amendment No. 93; Unit 3 - Amendment No. 76

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65673) The January 17, 1996, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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 application for amendment: February 16, 1996

Brief description of amendment: The amendment allows a one-time extension for the performance of the trip actuating device operational test for one of the safety injection manual initiation switches listed in Technical Specification Table 4.3-2, Item 1a.Date of issuance: March 11, 1996

Effective date: March 11, 1996

Amendment No. 63

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 7125). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 27, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated March 11, 1996

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: January 11, 1996

Brief description of amendments: The amendments revise the action statements and allowed outage time for inoperability of one channel and both channels of source range neutron flux instrumentation in Shutdown Modes 3, 4, and 5.

Date of issuance: March 15, 1996

Effective date: March 15, 1996

Amendment Nos.: 80, 80, 72, and 72

Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3509) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendments: November 14, 1995, as supplemented January 4, 1996 and February 29, 1996.

Brief description of amendments: The amendments revise the Technical Specifications to incorporate 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' Option B.

Date of issuance: March 11, 1996 Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 110 and 95

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

Date of initial notice in Federal Register: December 7, 1995 (60 FR 62896) The January 4, 1996, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan

Date of application for amendment: September 20, 1995, as supplemented December 18 and December 22, 1995.

Brief description of amendment: The amendment allows a one-time surveillance interval extension for certain 18-month surveillances listed in new Technical Specification Tables 4.0.2-1 and 4.0.2-2. Date of issuance: March 1, 1996

Effective date: March 1, 1996, with full implementation within 90 days.

Amendment No.: 106

Facility Operating License No. NPF-43. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58400). The December 18, 1995, letter corrected a typographical error on one of the proposed TS pages and provided a corrected Table of Contents page to reflect the addition of the new Tables. The December 22, 1995, letter provided additional information on the licensee's review of historical plant drift data. This information was within the scope of the original application and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

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

Date of application for amendments: November 10, 1995

Brief description of amendments: The amendments revise the Technical Specifications for containment systems to reflect the adoption of the requirements of 10 CFR Part 50, Appendix J, Option B, and the implementation of a performance-based containment leak-rate testing program at the Edwin I. Hatch Nuclear Plant, Units 1 and 2.

Date of issuance: March 6, 1996

Effective date: As of the date of issuance to be implemented within 90 days

Amendment Nos.: Unit 1 - 200 - Unit 2 - 141

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65679) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

Date of application for amendment: December 5, 1995

Brief description of amendment: The amendment revises the submittal date for the Annual Exposure Data Report bringing Oyster Creek into conference with 10 CFR 20.2206 and relaxes an overly restrictive administrative requirement.

Date of Issuance: March 4, 1996

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

Amendment No.: 183

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1629). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 4, 1996. No significant hazards consideration comments received: No.

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

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of application for amendment: December 14, 1995

Brief description of amendment: The amendment modifies Technical Specification 3.4.2, ''Flow Control Valves (FCVs),'' by deleting Surveillance Requirement (SR) 3.4.2.2, which required periodic verification that the average rate of movement of each reactor recirculation system FCV was limited to less than or equal to 11% per second in the opening and closing directions. Due to a plant modification, the requirement is not applicable.

Date of issuance: March 11, 1996

Effective date: March 11, 1996

Amendment No.: 103

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1630) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

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

Date of application for amendments: November 10, 1995 (AEP:NRC:0896X). This application superseded a request dated June 15, 1995 (AEP:NRC:0896V).

Brief description of amendments: The amendments change the 18-month emergency diesel generator surveillance test from a 24-hour run to an 8-hour run and add voltage and frequency measurement and power factor monitoring.

Date of issuance: March 11, 1996

Effective date: March 11, 1996, with full implementation within 45 days

Amendment Nos.: Unit 1 - 207, Unit 2 - 191

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65682) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 20, 1995, as supplemented December 19, 1995.

Brief description of amendments: The amendments relocate the fire protection program elements from the Technical Specifications and incorporate, by reference, the NRC-approved Fire Protection Program and major commitments, including the fire hazards analysis, into the Updated Final Safety Analysis Report. In addition, the amendments revise the operating licenses to include the NRC's standard fire protection license condition.

Date of issuance: March 11, 1996

Effective date: March 11, 1996, with full implementation within 180 days

Amendment Nos.: Unit 1 - 208, Unit 2 - 192

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications and the operating licenses.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47620). The December 19, 1995, supplement clarified the license conditions by providing specific approval dates for previous fire protection safety evaluations. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 29, 1995

Brief description of amendment: The amendment revises the Technical Specifications to extend the surveillance schedule from 18 months to each refueling interval (nominally 24 months) for specifications 4.6.4.2, 4.7.1.2.1.c, 4.7.3.b, 4.7.4.b,and 4.7.10.e. It also deletes specification 4.6.4.2.a and the phrase ''during shutdown'' from these specifications.Date of issuance: March 4, 1996

Effective date: As of the date of issuance, to be implemented within 90 days.

Amendment No.: 127

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58402) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Pacific Gas and Electric Company, Docket No. 50-275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California

Date of application for amendment: January 18, 1996

Brief description of amendment: The amendment revises the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit No. 1. TS 3.8.1.1, ''Electrical Power Systems - A.C. Sources - Operating,'' is revised to allow operation of Unit 1 in Mode 3 (Hot Standby) during installation of a replacement non-vital auxiliary transformer 11, for a one time extension of up to 48 hours beyond the 72 hours allowed by TS 3.8.1.1, Action Statement (a).

Date of issuance: March 8, 1996

Effective date: March 8, 1996

Amendment No.: Unit 1 - Amendment No. 111

Facility Operating License No. DPR-80: The amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 1, 1996 (61 FR 3737) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

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 application for amendments: December 27, 1995

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) 3/4.6.1.1, Containment Integrity; 3/4.6.1.2, Containment Leakage; 3/4.6.1.3, Containment Air Locks; 3/4.6.1.6, Containment Structural Integrity; 3/4.6.3, Containment Isolation Valves; their associated Bases; and adds Specification 6.8.4 j., Containment Leakage Rate Testing Program to implement the performance based leakage rate testing program as permitted by 10 CFR Part 50, Appendix J, rather than paraphrasing the requirements of the regulation. These changes will support the implementation of the performance based testing of Option B to Appendix J, for Type A, B, and C containment leakage rate testing and the appropriate rescheduling of testing.

Date of issuance: March 1, 1996

Effective date: March 1, 1996

Amendment Nos.: Unit 1 - Amendment No. 110; Unit 2 - Amendment No. 109

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3502) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

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

Date of application for amendment: April 13, 1994, as supplemented December 6, 1995

Brief description of amendment: The proposed changes revise the Quality Assurance audit frequencies in the Hope Creek Technical Specifications. These revisions will permit an audit frequency based on performance and transfer subsequent control over the audit program to the Updated Final Safety Analysis Report.

Date of issuance: March 11, 1996

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

Amendment No.: 95

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 8, 1994 (59 FR 29633) The December 6, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the original Federal Register notice.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 1996.No significant hazards consideration comments received: No

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

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: April 13, 1994, as supplemented December 6, 1995.

Brief description of amendments: The proposed changes revise the Quality Assurance audit frequencies in the Salem Unit Nos. 1 and 2 Technical Specifications. These revisions will permit an audit frequency based on performance and transfer subsequent control over the audit program to the Updated Final Safety Analysis Report.

Date of issuance: March 11, 1996

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

Amendment Nos. 181 and 162

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

Date of initial notice in Federal Register: June 8, 1994 (59 FR 29633) The December 6, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the original Federal Register notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 1996No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

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

Date of application for amendments: December 8, 1995 (TS 93-09)

Brief description of amendments: The amendments revise the setpoints and time delays for the auxiliary feedwater loss-of-power and the 6.9-kilovolt shutdown board loss-of-voltage and degraded voltage instruments.

Date of issuance: March 1, 1996

Effective date: March 1, 1996

Amendment Nos.: 219 and 209

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

Date of initial notice in Federal Register: January 3, 1996 (61 FR 181) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1996.No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

Date of application for amendments: January 4, 1996 (TS 95-22)

Brief description of amendments: The amendments change the surveillance test frequency specified for the functional tests of the containment, fuel storage pool, and control room radiation monitors from monthly to quarterly.

Date of issuance: March 4, 1996

Effective date: March 4, 1996

Amendment Nos.: 220 and 210

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

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3503)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated Macrh 4, 1996.No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

Date of application for amendment: January 16, 1996, and supplement dated March 1, 1996

Brief description of amendment: This amendment approves that part of the request that defers the drywell bypass leakage test during the current refueling outage. The remainder of the licensee's request is still under NRC staff review.

Date of issuance: March 8, 1996

Effective date: March 8, 1996

Amendment No. 82

Facility Operating License No. NPF-58: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 2, 1996 (61 FR 3951) The March 1, 1996, supplemental letter was clarifying in nature and did not affect the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: December 9, 1994, as supplemented by letters dated September 13, 1995, and February 9, 1996.

Brief description of amendment: The amendment revises Technical Specifications (TS) 4.3.2.2, TS 4.7.1.2.1, and the Bases for TS 3/4 7.1.2 to decrease the frequency of auxiliary feedwater pump testing, remove inconsistencies in testing requirements for the turbine-driven auxiliary feedwater pump, and clarify performance parameters in the TS Bases.

Date of issuance: March 11, 1996

Effective date: March 11, 1996, to be implemented within 30 days from the date of issuance.

Amendment No.: 108

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

Date of initial notice in Federal Register: February 1, 1995 (60 FR 6314). The September 13, 1995, and February 9, 1996, supplemental letters provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

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

Date of application for amendments: September 19, 1995

Brief description of amendments: The amendments revised the maximum allowable power range neutron flux high setpoints for operation with inoperable main steam safety valves.

Date of issuance: March 6, 1996

Effective date: March 6, 1996

Amendment Nos.: 199 and 180

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 25, 1995 (60 FR 54724) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 1996No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 22, 1995

Brief description of amendment: The amendment replaces the Technical Specification (TS) requirements associated with the boron dilution mitigation system (BDMS) with alarms, indicators, procedures and controls to allow proper resolution of potential boron dilution events.

Date of issuance: March 1, 1996

Effective date: March 1, 1996, to be implemented prior to the startup from the eighth refueling outage.

Amendment No.: 96

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3503) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1996.No significant hazards consideration comments received: No. Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

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

Date of amendment request: December 20, 1995, as supplemented by letter dated February 8, 1996.

Brief description of amendment: The amendment revises the Technical Specifications to reflect the approval of the use of 10 CFR Part 50, Appendix J, Option B for the Wolf Creek Generating Station containment leakage rate test program.

Date of issuance: March 1, 1996

Effective date: March 1, 1996, to be implemented prior to startup from the eighth refueling outage.

Amendment No.: 97

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3504) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1996.No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

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

Date of amendment request: December 13, 1995

Brief description of amendment: The amendment revises the minimum and maximum flow requirements for the centrifugal charging pumps (CCPs) and safety injection pumps (SIPs) specified in Technical Specification (TS) Surveillance Requirement 4.5.2.h. Specifically, the amendment (1) decreases the minimum limits on the sum of the injection line flow rates, excluding the highest flow rate, from 346 gallons per minute (gpm) to 330 gpm for the CCPs and from 459 gpm to 450 gpm for the SIPs, and (2) revises the maximum pump flow rate for the SIPs from 665 to 670 gpm, but retains the CCPs maximum pump flow rate at its current value of 556 gpm.Date of issuance: March 5, 1996

Effective date: March 5, 1996, to be implemented prior to startup from the eighth refueling outage.

Amendment No.: 98

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1639) The February 5, 1996, supplemental letter provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 1996.No significant hazards consideration comments received: No. Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a Federal Register notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By April 26, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

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

Date of application for amendment: March 6, 1996

Brief description of amendment: This amendment revises TS 3/4 5.2, ECCS SUBSYSTEMS - T_{avg} greater than or equal to 280°F by modifying Surveillance Requirement 4.5.2.b to defer venting of the Emergency Core Cooling System flow path which does not have manual venting capability until the tenth refueling outage.

Date of issuance: March 7, 1996

Effective date: March 7, 1996

Amendment No: 208

Facility Operating License No. NPF-3: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 7, 1996.

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

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

NRC Project Director: Gail H. Marcus

Dated at Rockville, Maryland, this 20th day of March 1996.

For the Nuclear Regulatory Commission

Steven A. Varga, Director,

Division of Reactor Projects - I/II,Office of Nuclear Reactor Regulation

[Doc. 96–7259 Filed 3–26–96; 8:45 am]

BILLING CODE 7590–01–F

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549

Extension:
Rule 11Ab2–1 and Form SIP
SEC File No. 270–23
OMB Control No. 3235–0043

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'') is publishing the following summary of collection for public comment.

Rule 11Ab2–1 and Form SIP establish the procedures by which a Securities Information Processor (''SIP'') files and amends its SIP registration form. The information filed with the Commission pursuant to Rule 11Ab2–1 and Form SIP is designed to provide the Commission with the information necessary to make the required findings under the Act before granting the SIP's application for registration. In addition, the requirement that a SIP file an amendment to correct any inaccurate information is designed to assure that the Commission has current, accurate information with respect to the SIP. This information is also made available to members of the public.

Only exclusive SIPs are required to register with the Commission. An exclusive SIP is a SIP which engages on an exclusive basis on behalf of any national securities exchange or registered securities association, or any national securities exchange or registered securities association which engages on an exclusive basis on its own behalf, in collecting, processing, or preparing for distribution or