4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the submittal may be viewed free of charge at the NRC Public Document Room, 2120 L Street NW, (lower level), Washington, DC. Members of the public who are in the Washington, DC, area can access this document via modem on the Public Document Bulletin Board (NRC's Advanced Copy Document Library), NRC subsystem at FedWorld, 703–321–3339. Members of the public who are located outside of the Washington, DC, area can dial FedWorld, 1–800–303–9672, or use the FedWorld Internet address: fedworld.gov (Telnet). The document will be available on the bulletin board for 30 days after the signature date of this notice. If assistance is needed in accessing the document, please contact the FedWorld help desk at 703–487–4608. Additional assistance in locating the document is available from the NRC Public Document Room, nationally at 1–800–397–4209, or within the Washington, DC, area at 202–634–3273.

Comments and questions should be directed to the OMB reviewer by December 6, 1996: Edward Michlovich, Office of Information and Regulatory Affairs (3150–0010), NEOB–10202, Office of Management and Budget, Washington, DC 20503.

Comments can also be submitted by phone at (202) 395–3084.

The NRC Clearance Officer is Brenda Jo. Shelton, (301) 415–7233.

Dated at Rockville, Maryland, this 7th day of October, 1996.

For the Nuclear Regulatory Commission.

Gerald F. Cranford,

Designated Senior, Official for Information Resources Management.

[FR Doc. 96–28505 Filed 11–5–96; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving 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 October 11, 1996, through October 25, 1996. The last biweekly notice was published on October 23, 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 Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By December 6, 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 20555-0001, 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 20555-0001, and to the attorney for the licensee.

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

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

Baltimore Gas and Electric Company, Docket No. 50-317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: October 3, 1996

Description of amendment request: The proposed amendment changes the provision for receiving, possessing and using byproducts, source and special nuclear material at Calvert Cliffs Unit 1.

Currently, Unit 1 is licensed under 10 CFR Part 30 to receive, possess, and use 100 millicuries of byproduct material for sample analysis or instrument calibration, 500 millicuries of byproduct material in the form of equipment; and 500 millicuries of Sodium-24 for steam turbine acceptance testing. In addition, Unit 1 is licensed to receive, possess and use 100 milligrams each of source or special nuclear material under 10 CFR Parts 40 and 70. Unit 2 is licensed under 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration or associated with radioactive apparatus or components. This proposed amendment would change the Unit 1 license to be consistent with the Unit 2 license by replacing license conditions 2.B.3 and 2.B.4 with the same wording as Unit 2's license condition 2.B.4.

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

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

Currently, Unit 1 is licensed under 10 CFR Part 30 to receive, possess, and use 100 millicuries of byproduct material for sample analysis or instrument calibration, 500 millicuries of byproduct material in the form of equipment; and 500 millicuries of Sodium-24 for steam turbine acceptance testing. Unit 1 is also licensed under 10 CFR parts 40 and 70 to receive, possess, and use 100 milligrams of source or special nuclear material. Unit 2 is licensed under 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration or associated with radioactive apparatus or components. This proposed amendment would change the Unit 1 license to be consistent with the Unit 2 license. The reason for this proposed change is that it is sometimes necessary to receive and use byproduct material, sources, or special nuclear material with different activity levels, and in different quantities than is specified by the Unit 1 license.

The current licenses for the two units allow radioactive materials to be accepted and used at Unit 2, although these same materials would not be acceptable for use at Unit 1. These byproduct, source, and special nuclear materials are used by the same people and for the same function in either unit. Training and procedures for handling radioactive material have been developed and used at both Units over the last 20 years. These procedures are adequate to control the acceptance and use of radioactive material at Unit 2 and, therefore, adequate to control radioactive material at Unit 1.

Receiving, possessing, and using byproduct, source, or special nuclear material is not related to accident conditions. Therefore, changing the Unit 1 license conditions to be the same as the Unit 2 license condition does not involve a significant increase in the probability or consequences of an accident.

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

Procedures and training governing the acceptance and use of radioactive materials are the same for both Unit 1 and Unit 2. These procedures will not be changed as a result of this license change. In addition, receiving, possessing, and using radioactive material is not related to accident conditions. Therefore, making the Unit 1 license the same as the Unit 2 license will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The margin of safety in this case is exposure to contaminated material or equipment. Exposure is controlled by adequate training and procedures. Radioactive material is received by personnel assigned to the Radiation Safety Section. These personnel are trained in receiving and shipping contaminated material. Once the material is onsite, it becomes the responsibility of the radiation protection staff who are trained in the handling of all levels of radioactive material. Training and procedures for handling radioactive materials have been developed and used over the 20-year life of the plant, and are currently deemed adequate for compliance with the Unit 2 license. Therefore, making the Unit 1 license the same as the Unit 2 license will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Calvert County Library, Prince Frederick, Maryland 20678.

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

NRC Project Director: S. Singh Bajwa, Acting Director

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: September 20, 1996

Description of amendment request: The proposed amendments would add a footnote to specification 4.3.1.B.4.A.10.a which refers to a letter that describes enhancements made to the Combustion Engineering sleeve installation process.

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 amendment continues to allow the Combustion Engineering sleeves to be used as an alternate tube repair method for Zion steam generators, along with the process enhancements which are described in the letter identified in the proposed Technical Specification note. The sleeve configuration, which was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and Section III of the ASME Code, is unaffected by the enhancements. Fatigue and stress analyses of the sleeved tube assemblies as described in the currently approved Topical Report, CEN-331-P, Revision 1-P, are unaffected by the enhancements.

Mechanical testing which has shown that the structural integrity of the sleeves under normal, faulted, and upset conditions is within the acceptable limits and is unaffected by the enhancements. Leakage rate testing for the tube sleeves which has demonstrated that primary to secondary leakage is not expected during any plant condition is unaffected by the enhancements. The consequences of leakage through the sleeved region of the tube, including the enhancements, is bounded by the existing steam generator tube rupture (SGTR) analysis included in the Zion Updated Final Safety Analysis Report.

The proposed Technical Specification change reflects enhancements to the installation and inspection process identified in Topical Report CEN-331-P, Revision 1-P, which is currently referenced in the Technical Specifications. These enhancements do not increase the probability or consequences of an accident previously evaluated. The enhancement which disallows the installation of the tube plugs made from Inconel 600 material was done so based upon industry information and is addressed by NRC Bulletin 89-01. The use of the Plus Point Probe, its associated data acquisition equipment, and improved visual inspection equipment, are conservative actions and improve the quality of the sleeving process. The use of the mechanical plug in lieu of the welded plug meets the established design requirements and is advantageous in the area of dose reduction, because of reduced time to install. Minor changes to the sleeve installation equipment as described in the Topical Report, represent equipment enhancements and do not alter the sleeve design or qualification testing.

The proposed Technical Specification change does not adversely impact any previously evaluated design basis accident. Installation of the sleeves, with the described enhancements, can be used to repair degraded tubes by returning the condition of the tubes to their original design basis condition for tube integrity and leak tightness during all plant conditions. Therefore, the currently approved sleeving process with the described enhancements will not increase the probability of occurrence of an accident previously evaluated.

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

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

The implementation of the enhancements to the proposed sleeving process will not affect the plant design basis. The current stress and fatigue analyses of the repair identified in Topical Report CEN-331-P, Revision 1-P, has shown the ASME Code and RG 1.121 allowable values are met and are unaffected by the described enhancements. The current sleeving design, with the described enhancements, will continue to maintain overall tube bundle structural integrity and leak tightness at a level consistent with that of the originally supplied tubing. Leak and mechanical testing of the sleeves, are unaffected by the proposed enhancements and continue to support the conclusions that the sleeve retains both structural integrity and leak tightness during all operating and accident conditions. Repair of a tube with a sleeve, utilizing the described enhancements, does not provide a mechanism that results in an accident outside of the area affected by the sleeve.

The described change to implement the cited enhancements will not create a new or different type of accident. The change only reflects enhancements to the currently approved installation/inspection process and, would not change or impact any hypothetical accident previously discussed. Use of improved Non-Destructive Examination, data acquisition and visual inspection equipment improves the quality of the sleeving process and has no negative effect on the margin of safety. The elimination of the use of the Inconel 600 plug also improves the margin of safety.

Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR analysis. The sleeve design, including described enhancements, does not affect any other component, or affect any location on the tube outside of the immediate area repaired.

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

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

The currently approved sleeving repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle to its original design basis condition. By implementing the described enhancements, the consistent quality of the upper sleeve weld has increased thereby reducing the potential for rework and reducing the potential for leaving a weld indication in service.

The proposed change does not involve a reduction in the margin of safety. The change reflects enhancements to the installation/inspection processes which are currently referenced in the Technical Specifications. These enhancements would not have any adverse effects on the previously evaluated design transient or accident analyses. The enhancements represent acceptable industry standards.

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

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

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

Date of amendment request: August 21, 1996

Description of amendment request: The proposed amendment will modify Containment Penetrations Nos. 53 and 65 design by modifying the design of instrumentation lines for Containment Vacuum Relief (CVR) system that pass through these containment penetrations. The proposed change will correct the error in previously docketed information that was used by NRC during licensing process.

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

The proposed change will not increase the probability of previously analyzed accidents. The proposed change seeks to clearly document the design and licensing bases for acceptance of the CVR sensing instrument lines. The proposed change to the monitoring lines will provide greater assurance that containment integrity will be maintained following a LOCA concurrent with a single active failure. The design change to the non-essential monitoring line will reduce the potential bypass leakage from penetrations 53 and 65 by adding a redundant automatic containment isolation valve on penetration 53 and isolating the non-essential instrument line on penetration 65. This design change can be performed at power without violating any license/regulatory requirements that ensure containment integrity is maintained.

There is no change in the function of the instrumentation. The only difference is that CVR-IDPT-5017B and C non-safety differential transmitters that monitor the CVR system will be sensing containment pressure from penetration 53. If the non-essential line coming from penetration 53 becomes inoperable, containment to annulus differential pressure can be obtained from alternate instrumentation. The essential sensing line that actuates the CVR system to protect containment within design vacuum pressure is not affected by the design change.

Adding a redundant automatic containment isolation valve in penetration 53's non-essential instrument line instead of the excess flow check valve and isolating the non-essential line in penetration 65's will significantly reduce the potential bypass leakage. The proposed change will credit the essential instrument lines as a closed system outside containment. The appropriate testing and acceptance criteria will be applied to ensure that any leakage associated with these potential bypass leakage paths, will not exceed the limits used in the Waterford 3 safety analysis or result in a significant increase in analyzed dose consequences. Therefore, the proposed change will not involve significant increase in the probability or consequences of any accident previously evaluated.

The proposed change will credit the essential sensing lines outside containment as a closed system and will not affect the plant or the manner in which the plant [is] operated.

The failure modes associated with containment isolation remain unchanged as a result of the design change to the non-essential monitoring lines. The function of the non-safety instrumentation is not affected. The only difference is that all of the non-safety instrumentation will be sensing containment pressure from penetration 53. However, if the non-essential line coming from penetration 53 becomes inoperable, containment pressure can be obtained from alternate instrumentation. Adding a redundant automatic containment isolation valve in series with CVR 401A in the non-essential instrument line ensures containment isolation following a LOCA with a concurrent a single active failure. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of a redundant automatic containment isolation valve in series with CVR 401A in the non-essential instrument line breaching penetration 53 ensures containment isolation postulating a single active failure on a Containment Isolation Actuation Signal (CIAS). While this modification is performed, administrative controls will require containment integrity to be maintained by a seismic Category 1, ASME Section III, Class 2, passive containment isolation device.

The essential CVR instrument sensing lines form a seismically qualified, closed system outside containment which is designed for pressure equal to or greater than containment. The instrument cabinets C-3A(B) are seismic Category I and safety related. The instruments are Safety Class 1E and have a static pressure rating of 1000 psig. These lines meet the criteria of BTP CSB 6-3 for crediting a closed system as a leakage boundary to preclude bypass leakage by being designed, fabricated, erected, and tested to standards commensurate with the safety function to be performed. The proposed change will apply the appropriate testing and acceptance criteria to ensure that any leakage associated with these potential bypass leakage paths, will not exceed the limits used in the Waterford 3 safety analysis or result in a significant increase in analyzed dose consequences. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 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: October 4, 1996 (TSCR No. 250)

Description of amendment request: The proposed Technical Specification (TS) change reflects a change in the Safety Limit Minimum Critical Power Ratio (SLMCPR) and as a result, a change in the operating Minimum Critical Power Ratio limit.

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

The derivation of the revised SLMCPR for Oyster Creek for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using NRC-approved methods. Additionally, interim implementing procedures, which incorporate cycle-specific parameters, have been used. Based on the use of these calculations, the revised SLMCPR will not increase the probability or consequences of an accident.

The basis of the MCPR Safety Limit calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPR preserves the existing margin to transition boiling and fuel damage in the event of a postulated accident. The probability of fuel damage is not increased.

Revising the operating MCPR limit for stability will ensure that adequate margin is retained to the SLMCPR.

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

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

The MCPR Safety Limit is a Technical Specification numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. The stability MCPR limit ensures an adequate operating MCPR margin to the SLMCPR. These revised limits cannot create the possibility of any new type of accident. The new SLMCPR has been calculated using NRC-approved methods. Additionally, interim procedures, which incorporate cycle-specific parameters, have been used. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

The margin of safety as defined in the TS Bases will remain the same. The new SLMCPR is calculated using NRC-approved methods which are in accordance with the current fuel design and licensing criteria. Additionally, interim implementing procedures, which incorporate cycle-specific parameters, have been used. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving fuel cladding integrity. The revised stability MCPR limit retains the existing margin to the SLMCPR. Therefore, the proposed TS change does not involve a reduction in a margin of safety.

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

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: October 10, 1996 (TSCR No. 203)

Description of amendment request: The proposed Technical Specification revision will extend the instrumentation surveillances for Condenser Low Vacuum, High Temperature Main Steamline Tunnel, Recirculation Flow, and Reactor Coolant Leakage. Additionally, the change will extend the equipment tests/operability checks for Containment Vent and Purge Isolation, Electromagnetic Relief Valve Operability, and Drywell to Torus Leakage Test. The above change extensions conform with the 24 month refueling interval.

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

The proposed amendment extends the period between successive refueling interval surveillance(s) to once every 24 months for those surveillance(s) evaluated herein. The proposed surveillance interval changes do not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operations. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed change to the surveillance interval. Assurance of system and equipment availability is maintained. This change does not involve any change to system or equipment configuration. Therefore, this change does not increase the probability of occurrence or the consequences of an accident previously evaluated.

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

The proposed amendment extends the period between successive refueling interval surveillance(s) to once every 24 months for those surveillance(s) evaluated herein. The proposed surveillance interval changes do not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operation. This change does not involve any change to system or equipment configuration. Therefore, this change is unrelated to the possibility of creating a new or different kind of accident from any previously evaluated.

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

The proposed amendment extends the period between successive refueling interval surveillance(s) to once every 24 months (+/-25% or 30 months) for the surveillances evaluated herein. The proposed surveillance interval changes do not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operation. The reliability of systems and components is not degraded by the proposed change to the surveillance interval. Assurance of system and equipment availability is maintained. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment 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: 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 application for amendment: October 10, 1996 (TSCR No. 243)

Description of amendment: The proposed Technical Specification (TS) will change the trip setting for TS Table 3.1.1 Item G.3, Automatic Depressurization System (ADS) by clarification of the functional requirement to provide an interlock permissive which ensures that a source of cooling water is available via the Core Spray System prior to depressurization. This will be accomplished by replacing the present interlock description ''AC Voltage'' with core spray booster pump differential pressure, as the permissive required for initiation of ADS. A corresponding surveillance requirement is being added to TS Table 4.1.1 which reflects the need to test and calibrate the core spray booster pump differential pressure switches pursuant to existing plant procedures. Additionally, allowed outage time (AOT) is addressed in the footnote ''i'' for the differential pressure switches based upon the currently designed ADS logic trains and footnote ''h'' to parallel the ''Low-Low Reactor Water Level'' and ''High Drywell Pressure'' AOTs associated with 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 facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

The implementation of this TSCR does not involve an increase in the probability of occurrence or the consequences of an accident previously evaluated, as no plant modifications are proposed by the change request, and no changes in instrument set or reset setpoints are required in order to implement the change. This change serves to clarify and to incorporate the ''as-built'' ADS system logic parameter (core spray booster pump differential pressure) as the functional permissive required for initiation of ADS. This ''interlock'' permissive compares closely with that of the BWR [boiling-water reactor] STS [Standard Technical Specifications] requirement to monitor core spray discharge pressure for initiation of ADS. In addition, the AOTs for the ADS initiation signals are being revised to align with the AOTs provided for such signals in the STS. The performance and function of the Automatic Depressurization System is unchanged by this request. However, by implementation of the change the specific functions of the ADS as-built d/p permissives would then be clearly identified in and controlled by T.S. Table 3.1.1, ''Protective Instrumentation Requirements,'' including the associated surveillance requirements as shown on the revised T.S. Table 4.1.1.

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

The implementation of this TSCR does not impact upon the operation of the facility, and would not create the possibility of a new or different kind of accident from any previously evaluated because no plant modifications are proposed by this change request, and no changes in instrument set or reset setpoints are required in order to implement the change. This change clarifies the technical specifications by incorporating the ''as-built'' ADS system logic parameter (core spray booster pump differential pressure) as the functional permissive required for initiation of ADS. This ''interlock'' permissive compares closely with that of the BWR STS requirement to monitor core spay discharge pressure. The revised AOTs for ADS initiation signals are also being changed to conform with those allowed by and provided in the STS. The performance and function of the Automatic Depressurization System (ADS) is unchanged by this request.

OC plant surveillance procedures for both ADS and the Core Spray system presently incorporate the calibration requirements and both the set and reset setpoints calculated for the core spray booster pump d/p switch permissive to the ADS initiation logic. Hence, a new or different kind of accident from any previously evaluated is not created.

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

The implementation of this TSCR does not involve a reduction in the margin of safety for operation of the ADS or the Core Spray system. The Technical Specification Bases which presently define the margin of safety are not impacted as the core spray booster pump d/p ''interlock'' permissives are not described in the specifications for ''Protective Instrumentation Requirements'' or its surveillance requirements. In addition, the margin of safety for ADS initiation is not reduced by this TSCR because the required system response is not affected by the proposed changes as no plant modifications are required which could create a potential impact upon the margins of safety previously established.

The revision of AOTs associated with ADS actuation signals by extension form 72 hours to 4 days is consistent with that presently provided in the STS. This does not decrease the margin of safety associated with availability of ADS as placement of the initiation signals into the ''tripped condition'' maintains the operability of the ADS trip systems while in the automatic mode. Additionally, the Bases for STS Specification 3.1 provides justifications for AOTs using the GE [General Electric] reliability analyses referenced therein and therefore 4 days is both justified and conservative. The margin of safety with respect to the instrument channels ability to perform its intended actuation function is not impacted; therefore, there is no reduction in the margin of safety.

Lastly, the surveillance frequency for the new surveillance interval created on Table 4.1.1 for the d/p [s]witches is consistent with that established in Reference 2 of the Bases for Technical Specification 4.1. Therefore, there is no reduction in the margin of safety as a result of this change request.

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

Local Public Document Room location: 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

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

Date of amendment request: October 17, 1996

Description of amendment request: The proposed amendment would revise Facility Operating License NPF-62 to acknowledge the transfer of Soyland Power Cooperative's 13.21% minority ownership interest in the Clinton Power Station to Illinova Power Marketing, Inc., the unregulated power marketing affiliate of Illinois Power, and a wholly owned subsidiary of Illinova Corporation.

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

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because it merely revises the Operating License to indicate the transfer of a minority ownership interest to the corporate parent of the majority owner and licensee. This proposed amendment represents an administrative rather than operational change and, therefore, has no impact on accidents previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because Illinois Power will continue to be the operator of Clinton Power Station, and further, there will be no change to the plant's physical configuration or operating philosophy as a result of this proposed amendment.

3. The proposed amendment does not involve a significant reduction in the margin of safety because it is only an administrative change and will have no impact on any margin of safety related to the design or operation of the facility.

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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

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

Date of amendment request: September 13, 1996

Description of amendment request: The proposed amendment would revise the Maine Yankee containment testing technical specification (TS 4.4) to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program'' dated September 1995.

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. This amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the proposed changes to the Technical Specification do not affect the assumption, parameters or results of any FSAR accident analysis.[...] These changes potentially result in a minor increase in the consequences of an accident previously evaluated due to the increased testing intervals. However, the proposed changes do not result in an increase in the probability of an accident previously identified since the containment system is used for mitigation purposes only. The changes are also expected to result in increased attention to components with poor leakage test history as part of the performance-based nature of Option B such that the marginally increased consequences from the expanded testing intervals may be further reduced or negated. The addition of the ''...[as modified by approved] exemptions'' phrase is an administrative change. Any specific exemptions from the requirements of Appendix J will continue to require a submittal under 10 CFR 50.12 and subsequent review and approval by the NRC prior to implementation. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of Maine Yankee in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) nor alter the function of the containment system. The changes only provide for additional time between leakage tests and an increase in the test pressure value equal to the containment design pressure which bounds the containment peak accident pressure. Thus, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of Maine Yankee in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The changes are expected to result in an increased focus on components demonstrating poor leakage test history without excessive testing of components which continue to demonstrate good test history. Therefore, these 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:* 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 A. Zwolinski

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

Date of amendment request: September 13, 1996, as supplemented September 25, 1996

Description of amendment request: The proposed amendment would revise TS 5.5.B to eliminate references to the Vice President (YNSD) and designate the President, Maine Yankee, as the responsible official for matters related to the composition, review and audit responsibilities, authority and recordkeeping responsibilities of the Nuclear Safety Audit and Review (NSAR) Committee. Minor editorial changes are also proposed.

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

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

The proposed change is administrative in nature and will not have a direct effect on the physical plant or the maintenance of the physical plant. The audit and review functions of the NSAR Committee will continue to be required. The proposed changes will not, of themselves, decrease the effectiveness of these functions. This authority and responsibility realignment will continue to assure that NSAR Committee has direct access to a level of management necessary to perform their audit and review functions.

Since, the proposed change will not adversely effect the audit and review functions of the NSARC and since the proposed change will not have a direct effect on the physical plant or maintenance of the physical plant, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is administrative in nature and does not introduce any new structures, systems, or components into the plant design. This change continues to ensure that the NSAR Committee reports to a management level such that there is sufficient authority and organizational freedom to execute their audit and review functions. Consequently, an unbiased oversight of the programs and procedures is not compromised by this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change realigns the authority and responsibility relationship of the NSAR Committee. The NSAR Committee will continue to maintain effective oversight of programs and procedures. The proposed change will continue to ensure that the NSAR Committee is sufficiently independent from cost and schedule when opposed to safety considerations. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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 A. Zwolinski

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

Date of amendment request: July 2, 1996

Description of amendment request: The proposed amendment incorporates limiting conditions for operation and surveillance requirements for the safety/relief valve (SRV) electrical lift design modification. The proposed amendment also makes clarification and editorial changes, as well as revising the associated Bases section.

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, NNECO has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration (SHC) since the proposed change satisfies the criteria in 10 CFR 50.92(c). That is, the proposed change does not:

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

The safety relief valves are considered for two analyzed accidents, an overpressure transient (such as MSIV [main steam isolation valve] closure with flux scram) and an inadvertent SRV opening.

The new technical specifications do not affect normal operation, therefore, they cannot increase the probability of an overpressure event. Since the mechanical function will not be affected by the new equipment, the new LCOs [limiting conditions for operation], or the new surveillance requirements, there is no adverse affect on the consequences of an overpressure event. The SRVs will be expected to lift mechanically. If they do not open at the design setpoints, the electrical actuation, which has the same setpoints, will cause the valves to open less than 400 milliseconds later.

Sufficient redundancy and diversity is established for the electrical lift by the use of two sensors in a two-out-of-two-taken-once configuration. Therefore, the failure of any single component cannot result in an inadvertent opening of an SRV. The only proposed surveillance performed while at power is the daily instrument check. This surveillance does not require the manipulation of any controls and, as such, cannot affect the probability of an accident.

Therefore, based on the above, the proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of any previously evaluated accident.

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

None of the proposed new LCOs or surveillance requirements has a potential for creating a new or different kind of accident. Expanding the LCO and surveillance requirements to address both the mechanical actuation and the pressure sensor lift does not change the type of action that these valves are expected to perform, nor does it change the initial ''as-left'' requirements for the valves. Plant operating parameters have also not changed.

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

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

The margin of safety previously analyzed for the SRVs was based on the current nominal setpoints and allowable percent drift. The electrical lift system improves the confidence that the SRVs will lift within the specified range. The setpoint uncertainty of the electrical lift system is similar to the drift allowed for the mechanical lift in the Technical Specifications. All existing functions that may actuate the SRVs (safety, manual, or automatic lift) remain unaffected. The design of the pressure transmitters, combined with the logic configuration, minimizes the possibility of inadvertently opening the SRVs.

Therefore, this change has 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, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

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

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: March 29, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Surveillance Requirement (SR) 4.5.1.d.2.b to delete the requirement to perform in-situ functional testing of the Automatic Depressurization System (ADS) valves once every 24-months as part of start-up testing activities.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems, or components (SSC). The ADS will continue to function as designed. The ADS is an Emergency Core Cooling System (ECCS) designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. The ADS utilizes five (5) of the 14 main steam line SRVs as the primary method for depressurizing the reactor pressure vessel to permit low pressure core cooling capability in the event of a small break Loss-of-Coolant-Accident (LOCA) if the high pressure cooling systems (i.e., High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems) fail to maintain adequate reactor vessel water level.

Deleting the TS SR to perform the in-situ testing of the ADS/SRVs during start-up, as proposed, should reduce the probability of an inadvertent opening of an SRV as discussed in Section 15.1.4 of the LGS Updated Final Safety Analysis Report (UFSAR) since deleting this testing requirement will eliminate a known initiator of SRV pilot leakage and subsequent erosion. This proposed TS change will have a tendency to increase, rather than decrease, the reliability of the ADS/SRVs by eliminating the in-situ ADS functional start-up testing. The probability of the ADS/SRVs to open on demand has been demonstrated to be extremely high and is not measurably improved through the in-situ ADS functional start-up testing.

This proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety. Alternate testing methods at LGS, Units 1 and 2, and at the off-site test facility, adequately demonstrate proper ADS valve operation and assure that the valves will continue to function as designed. Existing surveillance testing and inspections of the ADS/SRVs at LGS verify that the ADS initiation logic, solenoid valve operation, pneumatic gas supply integrity and air operator assembly (including pilot rod) will operate as designed. Offsite testing verifies pilot disc operation, setpoint calibration and main valve disc operation.

Deleting the in-situ testing requirement, as proposed, will reduce the probability of inflating SRV leakage which should reduce the probability of an inadvertent SRV opening. It has been documented throughout the BWR industry that pilot disc leakage leads to pilot disc and rod erosion, which can ultimately result in an inadvertent opening of an SRV. Therefore, any SRV pilot leakage that can be eliminated would reduce the probability of occurrence of a malfunction of that SRV.

Deleting the ADS/SRV in-situ functional test will in no way increase any consequences of a malfunction of plant equipment important to safety. The consequences of a malfunction of an ADS/SRV as discussed in the LGS UFSAR remain unchanged.

In addition, eliminating a known initiator of SRV leakage, as proposed in this TS change, would help to reduce operator workarounds in the form of suppression pool cooling and letdown operation activities. As a result, this will reduce the unnecessary operation of the Residual Heat Removal (RHR) and Residual Heat Removal Service Water (RHRSW) systems.

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

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

This proposed TS change does not involve any physical changes to plant SSC. The design and operation of the ADS/SRVs is not changed from that currently described in the Safety Analysis Report (SAR). The ADS will continue to function as designed to mitigate the consequences of an accident. No changes of any kind are being made to the valves, auxiliary components, or ADS logic. Deleting the requirement to perform the ADS in-situ functional test during plant start-up as proposed in this TS Change Request reduces the likelihood of a SRV developing a leak and degrading throughout the subsequent operating cycle. There is no possibility that implementing this proposed TS change would create a different type of malfunction to the ADS/SRVs than any previously evaluated.

Eliminating the requirement to perform the in-situ testing of the ADS/SRVs during start-up activities, does not create a new or different type of accident than any previously evaluated. There is no accident scenario associated with testing the ADS/SRVs other than the inadvertent opening of a relief valve which is currently discussed in Section 15.1.4 of the LGS UFSAR. This proposed TS change does not alter the conclusions described in the UFSAR regarding an inadvertent opening of an SRV. No new or different type of accident will be created as a result of this proposed TS change.

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

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

The proposed TS change does not involve any physical changes to plant SSC. The design and functional requirements of the ADS will not change. The ADS will still function as designed to mitigate the consequences of an accident.

This proposed TS change involves deleting the requirement to perform in-situ functional testing of the ADS/SRVs during start-up activities. This testing imposes an unnecessary challenge on the ADS/SRVs and has been linked to SRV degradation (e.g., pilot valve and/or main valve leakage). This proposed TS change should reduce SRV leakage and improve ADS/SRV reliability by reducing the potential for spurious SRV actuation. The LGS TS Bases do not identify specific testing requirements for ADS. ADS operability can be readily demonstrated with extremely high confidence by the existing additional surveillance tests and inspections performed for the ADS. There will be no reduction in any margin of safety resulting from this proposed TS change.

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

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

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: August 8, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Sections 3/4.3.1, ''Reactor Protection System Instrumentation,'' 3/4.3.2, ''Isolation Actuation Instrumentation,'' 3/4.3.3, ''Emergency Core Cooling System Actuation Instrumentation,'' and the associated TS Bases Sections 3/4.3.1 and 3/4.3.2 to eliminate selected response time testing requirements.

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

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

The proposed TS changes do not make any physical alterations or modifications to the plant systems or equipment. The proposed changes do not affect the capability of the associated systems to perform their intended functions within their required response times, nor do the proposed changes adversely impact the operation of any plant equipment. The affected plant systems will continue to function as designed. Elimination of the response time testing requirements as proposed by this TS change for selected components in RPS Instrumentation, Isolation Actuation System Instrumentation, and ECCS Actuation Instrumentation will not adversely affect the operation of these components.

The supporting analysis provided in NEDO-32291, demonstrates that response time testing is redundant to other TS required testing. NEDO-32291 demonstrated that these other required tests (i.e., channel checks, channel calibrations, channel functional tests, and logic system functional tests), in conjunction with actions taken in response to NRC Bulletin 90-01 and NRCB 90-01, Supplement 1, are sufficient to identify failure modes or degradation in instrument response times, and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by other TS required testing. The continued application of other existing TS required testing such as channel checks, channel calibrations, channel functional tests, and logic system functional tests, ensures that the response times for these systems will be maintained within the acceptance limits. The capability of these systems to perform their intended functions within their required response times is not adversely impacted by this proposed TS change. NEDO-32291 evaluated the potential failure modes of the affected instrumentation loops which could impact the instrument loop response times. Industry operating experience was also reviewed to identify failures that affect response times and how they are detected. The failure modes identified were evaluated to determine if other TS required surveillances and actions taken in response to NRC Bulletin 90-01, and NRCB 90-01, Supplement 1, would detect any effects on response time. There are no failures [sic] [failure] modes identified that can be detected by response time testing that cannot also be detected by other TS required testing.

PECO Energy has confirmed the applicability of the generic evaluation provided in NEDO-32291 to LGS, Units 1 and 2. By letter dated December 28, 1994, the NRC concluded that response time testing can be eliminated from the TS for the selected instrumentation identified in NEDO-32291, with certain provisions, and that NEDO-32291 can be referenced in license amendment requests.

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

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

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed changes apply only to the testing requirements for the selected components involved and do not result in any physical modifications to these components, or to other plant system components. Elimination of the response time testing requirements as proposed by this TS change for selected components in RPS Instrumentation, Isolation Actuation System Instrumentation, and ECCS Actuation Instrumentation will not adversely affect the operation of these components. These components will continue to function as designed. Consequently, no new failure modes are introduced as a result of the proposed TS changes.

Eliminating the response time testing requirements as proposed, does not create a new or different type of accident than any previously evaluated. No new or different type of accident will be created as a result of this proposed TS change.

NEDO-32291 demonstrates that other required tests (i.e., channel checks, channel calibrations, channel functional tests, and logic system functional tests), in conjunction with actions taken in response to NRC Bulletin 90-01 and NRCB 90-01, Supplement

1, are sufficient to identify failure modes or degradation in instrument response times, and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by other TS required testing, and therefore, response time testing for the selected components is redundant to the other TS required testing.

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

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

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed TS changes do not affect the capability of the associated systems or equipment from performing their intended functions. The systems involved will continue to respond within their allowed response times. Elimination of the response time testing requirements are based on the evaluation provided in NEDO-32291 which demonstrates that response time degradation can be detected by other TS required testing. The evaluation concluded that other TS required tests (i.e., channel checks, channel calibrations, channel functional tests, and logic system functional tests), in conjunction with actions taken in response to NRC Bulletin 90-01 and NRCB 90-01, Supplement 1, are sufficient to identify failure modes or degradation in instrument response times, and ensure operation of the associated systems within acceptable limits.

In addition, although not specifically evaluated, the proposed TS changes will provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing the potential for safety system actuations, reducing plant operating and shutdown risk, limiting radiation exposure to plant personnel, and eliminating the diversion of key personnel to conduct unnecessary testing. Therefore, PECO Energy considers that the proposed TS changes will result in an overall increase in the margin of safety and that the changes do not constitute an unreviewed safety question.

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

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

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: August 1, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Section 3/4.4.6 (i.e., Figure 3.4.6.1-1) to reflect the addition of two hydrotest curves, effective for 6.5 and 8.5 Effective Full Power Years (EFPY), to the existing Pressure-Temperature Operating Limit (PTOL) curves for LGS Unit 2.

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

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

The proposed Technical Specification (TS) change includes Pressure-Temperature Operating Limit (PTOL) curves which were conservatively generated in accordance with the fracture toughness requirements of 10CFR50, Appendix G. The Adjusted Reference Temperatures to the initial nil ductility reference temperatures (RTNDT) used to evaluate the pressure/temperature limits for the beltline materials were based on Regulatory Guide 1.99, Revision 2. Future analyses of the Reactor Pressure Vessel (RPV) surveillance capsule contents and future revisions to the PTOL curve as required, ensure that the reactor pressure boundary will behave in a non-brittle manner during plant testing, startup, and operation throughout the life of the plant. The current schedule for removal of the surveillance specimens from Limerick Generating Station (LGS) Unit 2 RPV is during 2R05. The proposed change does not impact the existing PTOL curves for 10 Effective Full Power Years (EFPY), currently shown in the LGS Unit 2 TS. The proposed change only provides additional information (i.e., two new curves) related to the RPV condition following 6.5 and 8.5 EFPY, in order to facilitate hydrostatic testing performed after 2R04 and 2R05, respectively. The added PTOL curves are established in compliance with the methodology used to calculate the predicted irradiation effects on vessel beltline materials as documented in the LGS Updated Final Safety Analysis Report (UFSAR). There are no physical changes to the plant being introduced by the added PTOL curves.

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

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

The proposed Technical Specification (TS) change includes Pressure-Temperature Operating Limit (PTOL) curves which were conservatively generated in accordance with the fracture toughness requirements of 10CFR50, Appendix G. The Adjusted Reference Temperatures to the initial nil ductility reference temperatures (RTNDT) used to evaluate the pressure/temperature limits for the beltline materials were based on Regulatory Guide 1.99, Revision 2. The proposed changes do not impact the existing PTOL curves for 10 Effective Full Power Years (EFPY), currently shown in the TS. They only provide additional information (i.e., two new curves) related to the reactor pressure vessel condition for 6.5 and 8.5 EFPY, in order to facilitate hydrostatic testing performed after 2R04 and 2R05, respectively. The added PTOL curves are established in compliance with the previous methodology used to calculate the predicted irradiation effects on vessel beltline materials as documented in the LGS [Updated Final Safety Analysis Report] UFSAR. The proposed TS change does not involve any physical changes to safety-related equipment.

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

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

The proposed change to Technical Specifications (TS) does not reduce the margin of safety as defined in the Bases for any TS. The added Pressure-Temperature Operating Limit (PTOL) curves for 6.5 and 8.5 Effective Full Power Years (EFPY) corresponding to 2R04 and 2R05, respectively, have been calculated in accordance with the existing methodology used to calculate the PTOL curves currently existing in the LGS Unit 2 TS (i.e., complying with the requirements of 10CFR50 Appendix G, and Regulatory Guide 1.99, Revision 2) and will more closely reflect the actual required reactor pressure vessel condition at the time in which the hydrotest is performed. Therefore, the margin of safety is not affected.

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

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

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: August 5, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Section 2.1 and its associated TS Basis to reflect the change in the Minimum Critical Power Ratio (MCPR) Safety Limit due to the plant specific evaluation performed by General Electric Co. (GE), for LGS Unit 2 Cycle 4.

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

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

The revised Minimum Critical Power Ratio (MCPR) Safety Limit for LGS Unit 2 Technical Specifications, and its use to determine cycle-specific thermal limits have been performed using NRC-approved methods within the existing design and licensing basis, and cannot increase the probability or severity of an accident.

The basis of the MCPR Safety Limit calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new MCPR Safety Limit preserves the existing margin to transition boiling and fuel damage in the event of a postulated accident. The probability of fuel damage is not increased.

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

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

The MCPR Safety Limit is a Technical Specification numerical value, designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. It cannot create the possibility of any new type of accident. The new Minimum Critical Power Ratio (MCPR) Safety Limit is calculated using NRC-approved methods and is based on LGS Unit 2 Cycle 4 specific inputs.

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

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

The margin of safety as defined in the TS Bases will remain the same. The new Minimum Critical Power Ratio (MCPR) Safety Limit is calculated using NRC approved methods which are in accordance with the current fuel design and licensing criteria. The MCPR Safety Limit remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

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

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendments request: September 30, 1996

Description of amendments request: The proposed amendments would revise Technical Specifications (TSs) 3/4.1.1, 3/4.1.3, 3.1.3.6, 3.2.1, 3/4.2.2, and 3.2.3 and associated Bases to remove certain cycle-specific parameter limits from the TSs and relocate them to the Core Operating Limits Report (COLR). These changes result from NRC Generic Letter (GL) 88-16, dated October 4, 1988, which provided guidance to licensees on requests for removal of the values of cycle-specific parameter limits from the TSs. The licensee's proposed amendments are consistent with the GL.

The COLR has been included in the Definitions section of the TSs. The definition notes that it is the unit-specific document that provides these limits for the current operating reload cycle. The values of these cycle-specific parameter limits are to be determined in accordance with TS 6.9.1.11. This TS requires that the core operating limits be determined for each reload cycle in accordance with the referenced NRC-approved methodology for these limits and consistent with the applicable limits of the safety analysis. The COLR shall be provided to the NRC upon issuance.In addition, the above TS changes would produce administrative changes to the TS Table of Contents.

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

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

The removal of cycle-specific core operating limits from the FNP [Farley Nuclear Plant] Technical Specifications has no influence or impact on the probability or consequences of a Design Basis Accident (DBA) occurrence. The cycle-specific core operating limits, although not in Technical Specifications, will be followed in the operation of FNP. The proposed amendment retains the same required actions to be taken when or if limits are exceeded as stipulated by current Technical Specifications. In addition, the associated surveillance requirements are not altered by the proposed changes.

Each accident analysis addressed in the FNP FSAR [Final Safety Analysis Report] will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC-approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed per requirements of 10 CFR 50.59, ensures that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As stated earlier, the removal of the cycle-specific variables has no influence or impact, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The cycle-specific variables are calculated using the NRC-approved methods and submitted to the NRC to allow the Staff to continue to trend the values of these limits. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is not affected by the removal of cycle-specific core operating limits from the Technical Specifications. The margin of safety presently provided by current Technical Specifications remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed amendment continues to require operation within the core limits, as obtained from the NRC-approved reload design methodologies. The required actions to be taken or if limits are violated remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC-approved documentation. In addition, each future reload involves a 10 CFR 50.59 safety review to assure that operation of FNP within the cycle-specific limits will not involve a significant reduction in [the] margin of safety. Therefore, the proposed changes are administrative in nature and do not impact the operation of FNP in a manner that involves a reduction to 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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

NRC Project Director: Herbert N. Berkow

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

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

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

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

Date of amendment request: August 29, 1996

Description of amendment request: The proposed amendment would modify the applicability requirements for certain radiation monitors so that the radiation monitors are required to be operable only when secondary containment integrity is required to be operable; delineate when secondary containment integrity is required; modify standby gas treatment operability requirements; make editorial corrections to clarify the configuration of the radiation monitors; and revise the associated Bases section.

Date of publication of individual notice in Federal Register: October 17, 1996 (61 FR 54242)

Expiration date of individual notice: November 18, 1996

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

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

Date of application for amendment: September 6, 1996

Brief description of amendment request: The proposed amendment would change Technical Specification (TS) requirements related to steam generator tubes to allow a laser-welded repair of Westinghouse hybrid expansion joint (HEJ) sleeved steam generator tubes. Date of individual notice in Federal Register: October 15, 1996 (61 FR 53769)

Expiration date of individual notice: November 14, 1996

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin

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: June 17, 1996

Brief description of amendments: The amendment modifies the technical specifications (TS) to change (1) the reference method for calculating dose conversion factors (DCFs) to be used in dose calculations, and (2) the upper and lower limits for operating pressurizer pressure to account for new instrument uncertainties and to reduce the allowed operating band.

Date of issuance: October 23, 1996

Effective date: October 23, 1996, to be implemented within 45 days of issuance

Amendment Nos.: Unit 1 - 109; Unit 2 - 101; Unit 3 - 81

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

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47963). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 23, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 9, 1995

Brief description of amendments: The amendments implement changes to radiological effluent Technical Specifications in accordance with Generic Letter 89-01 ''Implementation of Programmatic for Radiological Effluent Technical Specification in the Administrative Controls Section of the Technical Specifications and Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.''

Date of issuance: October 18, 1996

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

Amendment Nos.: 217 and 194

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

Date of initial notice in Federal Register: July 15, 1995 (60 FR 35062) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 18, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 1, 1996

Brief description of amendment: The proposed amendment will reflect the implementation of 10 CFR Part 50 Appendix J, Option B at the Pilgrim Nuclear Power Station.

Date of issuance: October 4, 1996

Effective date: October 4, 1996

Amendment No.: 167

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

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28606) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1996. No significant hazards consideration comments received: No

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

Carolina Power & Light Company, et al., Docket No. 50-325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: April 8, 1996, as supplemented on July 30, 1996, October 4, 1996, October 8, 1996, and October 16, 1996.

Brief description of amendment: The amendment changes the Technical Specifications to (1) reflect the use of a new type of fuel (GE13) and (2) modify the minimum critical power ratio safety limit and the standby liquid control system sodium pentaborate limits to accommodate the GE13 fuel.

Date of issuance: October 17, 1996

Effective date: October 17, 1996

Amendment No.: 182

Facility Operating License No. DPR-71: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42276) which superseded a Federal Register notice published on June 5, 1996 (61 FR 28607) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: January 5, 1996, as supplemented July 12, 1996

Brief description of amendment: The amendment revises the shutdown cooling (SDC) requirement to allow one train of the SDC system to be rendered inoperable for testing or maintenance provided that a filled refueling cavity is available to provide backup decay heat removal capability in the event that the operating train of SDC becomes inoperable.

Date of issuance: October 10, 1996

Effective date: October 10, 1996

Amendment No.: 173

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44348) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: August 8, 1996

Brief description of amendments: The amendments revise the Technical Specifications, Section 6.9.1.9, to reference updated or recently approved topical reports used to calculate cycle-specific limits contained in the Core Operating Limits Report.

Date of issuance: October 24, 1996

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

Amendment Nos.: 154 and 146

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47977) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 1996.No significant hazards consideration comments received: No

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

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2, Pope County, Arkansas

Date of amendment request: May 9, 1996

Brief description of amendments: The amendments revised the name from Arkansas Power & Light Company to Entergy Arkansas, Inc.

Date of issuance: October 23, 1996

Effective date: October 23, 1996

Amendment Nos.: 187 and 177

Facility Operating License Nos. DPR-51 and NPF-6. Amendments revised the Technical Specifications and the licenses.

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44357) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 23, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 8, 1996, as supplemented by letters dated July 18 and September 19, 1996

Brief description of amendment: The amendment modified the frequency requirements in Surveillance Requirement 3.6.1.3.5 of the Technical Specifications, on the leakage rate testing for each containment purge isolation valve with resilient seals, to place these purge valves on a performance basis in accordance with Appendix J of 10 CFR Part 50, as modified by any exemptions to Appendix J. In addition, the purge valves would be required to be leakage rate tested every 36 months with at least two pairs tested every 18 months and, if any purge valve fails to meet the leakage rate acceptance criterion, all remaining valves must be tested within 92 days (i.e., a quarter of a year) if not successfully tested within the previous 92 days.

Date if issuance: October 18, 1996

Effective date: October 18, 1996

Amendment No.: 128

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28614) The additional information contained in the supplemental letters dated July 18 and September 19, 1996, revised the proposed amendment in the application of May 8, 1996; however, the revisions were within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 20, 1996, as supplemented by the letter of September 11, 1996

Brief description of amendment: The amendment redefined the secondary containment boundary to allow the enclosure building to be inoperable during the upcoming refueling Outage 8 (RFO 8) scheduled to begin in October 1996. The amendment added a condition to the license that the enclosure building may be inoperable during core alterations and movement of non-recently irradiated fuel (i.e., fuel that has not occupied part of a critical reactor core for 12 days) during RFO 8 and the standby gas treatment (SGT) system may be unable to automatically start or achieve and maintain the required vacuum, provided the following conditions exist:

a. All dampers communicating between the auxiliary building and the enclosure building are closed.

b. The access door between the auxiliary building and the enclosure building is closed, except when the access opening is being used for entry and exit.

c. The SGT system is blocked from automatic initiation.

d. The SGT system is available for manual initiation or the actions for Limiting Condition for Operation 3.6.4.3 in the Technical Specifications for GGNS are complied with.

The non-recently irradiated fuel is spent fuel that has decayed at least 12 days after the reactor was shut down for refueling.

Date of issuance: October 18, 1996

Effective date: October 18, 1996

Amendment No: 129

Facility Operating License No. NPF-29. Amendment adds a condition to the license.

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37299) The additional information contained in the supplemental letter of September 11, 1996, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 1996.No significant hazards consideration comments received: No

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

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: February 22, 1996, and as supplemented by letters dated July 22 and September 20, 1996

Brief description of amendment: The amendment revises Clinton Power Station Technical Specification 3.4.11, ''Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits,'' to incorporate specific P/T limits for the bottom head region of the reactor vessel, separate and apart from the core beltline region of the reactor vessel.

Date of issuance: October 23, 1996

Effective date: October 23, 1996

Amendment No.: 109

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18169) The letters of July 22 and September 20, 1996, provided clarifying information and did not alter the staff's initial finding that the proposed changes involve no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 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

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

Date of application for amendment: November 21, 1995

Brief description of amendment: The amendment changes Technical Specification Section 5.2.2, ''Design Pressure and Temperature,'' to clarify that the reactor containment design temperature is an equilibrium liner temperature and not the air temperature. The supporting Technical Specification Bases is updated to reflect the change and to include the main steam line break accident, in addition to the loss-of-coolant accident, as the limiting events affecting the containment temperature and pressure.

Date of issuance: October 21, 1996

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

Amendment No.: 204

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65684) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 21, 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, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

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

Date of amendment requests: July 15, 1996, as supplemented by letters dated September 3, 1996, October 22, 1996, October 23, 1996, and August 23, 1996

Brief description of amendment: The amendment revises Technical Specifications (TS) Section 4.3.2 to allow the use of zircaloy or ZIRLO fuel cladding and to use depleted uranium as reactor fuel material. The amendment also changes TS Section 5.9.5 to add Westinghouse Topical Reports, WCAP-12610-P-A, ''VANTAGE + Fuel Assembly Report,'' and WCAP-13027-P, ''Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS,'' to the list of approved analytical methods for determining the core operating limits.

Date of issuance: October 25, 1996

Effective date: October 25, 1996

Amendment No.: 178

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40026) and August 30, 1996 (61 FR 45995). The September 3, 1996, October 22, 1996, and October 24, 1996, supplemental letters provided additional clarifying and correcting information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

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: June 7, 1996

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 by revising Technical Specifications 3/4.9.14.1, ''Spent Fuel Assembly Storage - Spent Fuel Pool Region 2,'' and TS 3/4.9.14.3, ''Spent Fuel Assembly Storage - Spent Fuel Pool Region 1,'' to allow storage of fuel assemblies in a checkerboard pattern in Region 2 of the spent fuel pool (SFP).

Date of issuance: October 25, 1996

Effective date: October 25, 1996, to be implemented within 30 days from date of issuance.

Amendment Nos.: Unit 1 - 116; Unit 2 - 114

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

Date of initial notice in Federal Register: September 25, 1996 (61 FR 50346) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 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 19, 1995, as supplemented by letter dated August 8, 1996.

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 to relocate Technical Specification (TS) 6.5, ''Review and Audit,'' 6.8, ''Procedures and Programs,'' Sections 6.8.1c., 6.8.1d., 6.8.2, and 6.8.3, in accordance with guidance in an NRC letter dated October 25, 1993, from William T. Russell to the chairpersons of industry owners groups and the Commission's Final Policy Statement on TS Improvements for Nuclear Power Reactors on relocation of TS that do not satisfy the retention criteria. As part of the relocation of TS 6.8.2, TS 6.1.1 would be revised to require that proposed tests, experiments, or modifications that affect nuclear safety be approved by the plant manager or his designee prior to implementation.

Date of issuance: October 25, 1996

Effective date: October 25, 1996, to be implemented within 90 days of issuance.

Amendment Nos.: Unit 1 - 117; Unit 2 - 115

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1633) The August 8, 1996, supplemental letter provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 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

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: May 20 and 28, 1996, as supplemented by letter dated July 25, 1996

Brief description of amendments: These amendments, for both units, add a reference to the ANF-B critical power correlation to Section 6.9.3.2 of the Technical Specifications (TSs); change the values of the minimum critical power ratio (MCPR) in TS Sections 2.1 and 3.4.1.1.2, and make appropriate Bases changes. For Unit 1 only, a reference to ABB licensing methodology report CENPD-300 (for lead use assemblies being used in the reactor core during the upcoming operating cycle) is added to Section 6.9.3.2.

Date of issuance: October 11, 1996

Effective date: For both units, as of date of issuance, to be implemented within 30 days.

Amendment Nos.: 161 and 132

Facility Operating License Nos. NPF-14 and NPF-22. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: Unit 2, August 28, 1996 (61 FR 44362); Unit 1, September 4, 1996 (61 FR 47529)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 11, 1996.No significant hazards consideration comments received: No

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

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: March 29, 1996, as supplemented July 12, 1996, and September 6, 1996.

Brief description of amendment: The proposed amendment would change the Indian Point 3 Technical Specifications (TSs) relating to minimum reactor coolant system (RCS) flow and maximum RCS average temperature to make these parameters consistent with an assumption of 100% helium release from the boron coating of the integral fuel burnable absorber rods. The proposed amendment would also add limits associated with Departure from Nucleate Boiling to the IP3 Technical Specifications TSs.

Date of issuance: October 22, 1996

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

Amendment No.: 170

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37301) August 14, 1996 (61 FR 42283)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 19, 1995, as supplemented by letters dated December 22, 1995, and March 26, 1996.

Brief description of amendments: These amendments modify Technical Specification (TS) 3.3.8, ''Containment Purge Isolation Signal (CPIS),'' and TS 3.3.9, ''Control Room Isolation Signal (CRIS).'' The revisions are needed to (1) support the upgrading or replacement of existing radiation monitoring system with state-of-the-art equipment that will provide for greater operational flexibility and reliability, and (2) incorporate minor editorial changes to improve clarity of these TS sections.

Date of issuance: October 8, 1996

Effective date: October 8, 1996, to be implemented within 30 days of date of issuance

Amendment Nos.: Unit 2 - Amendment No. 132; Unit 3 - Amendment No. 121

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49948). The December 22, 1995, and March 26, 1996, letters provided additional 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 October 8, 1996.No significant hazards consideration comments received: No.

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

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: March 29, 1996, as supplemented by letters dated June 27, August 29, and September 16, 1996.

Brief description of amendment: The amendment changes Technical Specification 3/4.4.6, ''Steam Generators'' and associated Bases to modify the steam generator repair limit to clarify that the appropriate method for determining serviceability for tubes with outside diameter stress corrosion cracking at the tube support plate is by a methodology that more reliably assesses structural integrity.

Date of issuance: October 11, 1996

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

Amendment No.: 115

Facility Operating License No. NPF-8: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25711) The June 27, August 29, and September 16, 1996, letters provided additional, clarifying information that did not change the scope of the March 29, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 11, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: April 22, 1996, as supplemented by letters dated May 3, July 15, August 7 and 30, and September 16, 1996

Brief description of amendment: The amendment changes reflect the implementation of a new F* criterion based on maintaining existing safety margins for steam generator tube structural integrity concurrent with allowances for nondestructive examination eddy current uncertainty.

Date of issuance: October 11, 1996

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

Amendment No.: 116

Facility Operating License No. NPF-8: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25713) The May 3, July 15, August 7 and 30, and September 16, 1996, letters provided clarifying information that did not change the scope of the April 22, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 11, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Tennessee Valley Authority, Docket No. 50-390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: June 29, 1996

Brief description of amendment: The amendment revises Technical Specification (TS) Section 5.2.2.f to delete the sentence, ''The Operations Manager shall hold or have held an SRO [Senior Reactor Operator] license on a similar unit.'' The revision also indicates that the Operations Superintendent will hold a valid SRO license on this unit.

Date of issuance: October 15, 1996

Effective date: Octber 15, 1996

Amendment No.: 4

Facility Operating License No. NPF-90: Amendment revises the TS.

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47983)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 15, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 31, 1996 (TXX-96432) as supplemented by letters dated August 23 and 27 (TXX-96447 and TXX-96451), and September 19, 1996 (TXX-96469).

Brief description of amendments: The amendments (1) change the acceptance values for amperes and voltage for the 18 month surveillance test of the battery chargers; (2) clarify the meaning of the term ''associated inverter'' used in the context of energizing 118-Volt AC Instrument Buses during MODES 1 through 6; and (3) delete the protection channel and the vital bus ratings for the 118-Volt AC Instrument Buses identified for MODES 1 through 4.

Date of issuance: October 22, 1996

Effective date: October 22, 1996

Amendment Nos.: Unit 1 - Amendment No. 53; Unit 2 - Amendment No. 39

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44363) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 22, 1996.No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 30th day of October 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 96-28372 Filed 11-5-96; 8:45 am]

BILLING CODE 7590-01-F

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (The Alpine Group, Inc., Common Stock, $0.10 Par Value) File No. 1–9078

October 31, 1996.

The Alpine Group, Inc. (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified security (''Security'') from listing and registration on the American Stock Exchange, Inc. (''Amex'').

The reasons alleged in the application for withdrawing the Security from listing and registration include the following:

According to the Company, the Board of Directors (the ''Board'') adopted a resolution as of September 27, 1996 to withdraw the Security from listing on the Amex and, instead, to list such Security on the New York Stock Exchange (''NYSE''). The decision of the Board on this matter followed an appropriate exploration of means to enhance stockholder value, and was based upon the belief that the listing of the Security on the NYSE will be more beneficial to its shareholders than continued listing on the Amex.

Any interested person may, on or before November 22, 1996, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549, facts bearing upon whether the application has been made in accordance with the rules of the exchanges and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.

Jonathan G. Katz,

Secretary.

[FR Doc. 96–28455 Filed 11–5–96; 8:45 am]

BILLING CODE 8010–01–M

[Rel. No. IC–22306; File No. 811–7796]

ILI Endeavor Variable Annuity Account

October 30, 1996.

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

ACTION: Notice of Application for an order under the Investment Company Act of 1940 (''1940 Act'').

APPLICANT: ILI Endeavor Variable Annuity Account.

RELEVANT 1940 ACT SECTION: Order requested under Section 8(f) of the 1940 Act.

SUMMARY OF APPLICATION: Applicant seeks an order declaring that it has ceased to be an investment company as defined by the 1940 Act.

FILING DATE: The application was filed on July 7, 1996.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Secretary of the SEC and serving Applicant with a copy of the request, in person or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on November 25, 1996, and should be accompanied by proof of service on Applicant in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the requestor's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the Secretary of the SEC.

ADDRESSES: Secretary, Securities and Exchange Commission, 450 5th Street, N.W., Washington, D.C. 20549. Applicant, Frank A. Camp, Esq., International Life Investors Insurance Company, 4333 Edgewood Road N.E., Cedar Rapids, Iowa 52499.

FOR FURTHER INFORMATION CONTACT: Patrice M. Pitts, Branch Chief, or Michael Koffler, Law Clerk, Office of Insurance Products (Division of Investment Management), at (202) 942–0670.

SUPPLEMENTARY INFORMATION: Following is a summary of the application; the complete application is available for a fee from the Public Reference Branch of the SEC.

Applicant's Representations

1. Applicant, a unit investment trust, is a separate account of International Life Investors Insurance Company (''ILI'') designed as a funding medium for variable annuity contracts (''Contracts''). On June 14, 1993, Applicant filed with the Commission a notification of registration as an investment company on Form N–8A, and a registration statement under Section 8(b) of the 1940 Act and under the Securities Act of 1933 (File No. 33–64414) registering an indefinite amount of securities (*i.e.*, the Contracts). The registration statement was declared effective, August 12, 1993, and Applicant began offering Contracts on August 12, 1993.

2. The boards of directors of ILI and AUSA Life Insurance Company (''AUSA Life'') authorized the adoption of an ''Assumption Reinsurance Agreement'' on September 27, 1994. Contractholders were given the right to reject the assumption of their Contracts by AUSA Life, as required by the law of the State of New York, via a solicitation dated December 1, 1994. No contractholders rejected the assumption of their Contracts pursuant to the terms of the solicitation.

3. The Assumption Reinsurance Agreement, dated as of December 31, 1994, providef for the transfer of the in force variable annuity business of ILI to AUSA Life, as of January 1, 1995. Effective January 1, 1995, ILI ceded and transferred to ASUA Life all variable insurance contracts issued by ILI in connection with its variable annuity business.