Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated November 22, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Dated at Rockville, Maryland, this 9th day of January 1997.

For the Nuclear Regulatory Commission.

George F. Wunder,

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

[FR Doc. 97–982 Filed 1–14–97; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 20, 1996, through January 3, 1997. The last biweekly notice was published on January 2, 1997 (62 FR 121).

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 February 14, 1997, 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.

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

Date of amendment request: November 4, 1996, as supplemented on December 4, 1996.

Description of amendment request: The proposed amendment would permit Byron, Unit 1, and Braidwood, Unit 1, to remove sheathing filler grease in the tendon sheathing for up to 35 tendons in advance of the steam generator replacement outages.

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 prestressing tendons are passive components that form part of the containment structure. As passive components, there are no tendon failure modes that could act as accident initiators or precursors.

Consequently, the proposed change to remove a portion of the tendon sheathing filler grease will not increase the probability of an accident previously evaluated.

The tendons, in their passive role, function to limit the consequences of accidents previously evaluated, and their continued integrity is important to the ability of the containment to mitigate design basis accidents. Structural degradation of the containment is a predictable process that can be monitored by a comprehensive containment tendon monitoring program as required by Technical Specification Surveillance Requirement 4.6.1.6. The monitoring program is based on proposed Revision 3 of Regulatory Guide 1.35, ''Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures,'' April 1979.

The tendon surveillances conducted at both Byron and Braidwood have consistently shown that structural integrity of the tendon system has been maintained, including adequate corrosion protection for the tendon wires and end anchorage components, and there has been no evidence of grease leakage from the tendon sheathings. While a number of below-grade hoop tendons have shown signs of water intrusion, the tendons that will have grease removed are above-grade and are not expected to experience water intrusion.

A review of domestic nuclear facility experience found cases where large grease voids existed for periods longer than requested under the proposed change without resultant corrosion in those tendon systems. A case where tendon wires removed from a decommissioned plant were exposed to an environment more severe than expected in a sealed tendon sheath did not show signs of corrosion. These experiences demonstrate the effectiveness of the initial corrosion protection systems applied to the tendons and the effectiveness of partial grease protection in the tendon sheathing.

Based on the above cases, it can be concluded that the removal of the filler grease (grease voids greater than 5 percent) from the tendon sheathing in up to thirty-five tendons for a limited period will not adversely affect the integrity of the tendons or the capability of the tendon system to fulfill its design basis function.

The removal process will only remove the grease not directly adhering to the tendons. The grease remaining will be adequate to protect the tendons during the relatively short period of partial grease removal. Therefore, no changes in the tendon properties would be expected, and the consequences of design basis accidents previously evaluated will not be affected by the proposed change.

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 only affects the tendon sheathing filler grease void limits of TSSR 4.6.1.6. No new equipment is being installed and no existing equipment is being modified. Operation with a grease void in excess of current requirements does not alter system configurations such that any new or different accidents can be initiated. Therefore, no new or different accident initiators or precursors are being introduced, and the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The margin of safety applicable to the proposed change is defined by the difference between the design pressure of the containment and the point at which the containment would actually fail. The design pressure of the containment is 50 psi. As a result of conservatism inherent in the design techniques and in the material selections made for the Byron and Braidwood containments, a substantial margin to failure exists in the containment. This margin is discussed in Subsection 3.8.1.8 of the Updated Final Safety Analysis Report. It is noted therein that the ultimate capacity of the concrete shell is 125 psi, corresponding to the initiation of yield in the hoop post-tensioning tendons in conjunction with yielding of the reinforcement near the mid-height of the containment wall.

It is also noted in Subsection 3.8.1.8 that the ultimate capacity of a containment electrical penetration is 108 psi. While this value is substantially greater than the 50 psi required of the design, it is lower than the 125 psi at which failure of the containment wall section would be predicted. Therefore, tendon strength is not the limiting factor in the margin of safety inherent in the containment.

As previously discussed, no degradation of the tendons is expected to occur as a result of the proposed TS change. Further, the tendon strength is not the limiting factor in the containment ultimate capacity, which is substantially greater than the requirement placed on the containment design by the plant design basis. Therefore, the proposed change will not reduce the margin of safety designed into Byron and Braidwood.

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

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

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: December 6, 1996

Description of amendment request: The proposed amendment would allow a single control rod to be moved when the plant is in HOT SHUTDOWN and COLD SHUTDOWN condition provided the one-rod-out interlock is OPERABLE and the reactor mode switch is in the refuel position.

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

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

This revision would allow a single control rod to be withdrawn under control of the reactor mode switch position one-rod-out interlock in OPERATIONAL MODES 3 or 4. This interlock is explicitly assumed in the safety analysis for control rod removal error during refueling. A prompt reactivity excursion could potentially result in fuel failure. The one-rod-out interlock, together with the requirements for adequate SHUTDOWN MARGIN (SDM), provides protection against prompt reactivity excursions by preventing withdrawal of more than one control rod and ensuring the core remains subcritical with any one control rod withdrawn. The addition of surveillance requirements for the one-rod-out interlock will assure the interlock is OPERABLE prior to withdrawal of a control rod in OPERATIONAL MODES 3 and 4. Although this change will increase the frequency of single control rod withdrawals in OPERATIONAL MODES 3 and 4, the probability of previously analyzed accidents, including control rod withdrawal error, is not affected because the same actions are required, although they are now conducted in different OPERATIONAL MODES.

The consequences of previously analyzed accidents in OPERATIONAL MODES 3 and 4 are not affected by this proposed change. The SDM requirements of TS 3.3.A assure the reactor is maintained subcritical when all control rods are fully inserted, without crediting the single control rod having the highest reactivity worth which is assumed to be fully withdrawn. The one-rod-out interlock of the reactor mode switch Refuel position permits only a single control rod to be withdrawn. The proposed change will not affect the potential for attaining criticality in OPERATIONAL MODES 3 and 4 or effect the initial conditions assumed in any design basis accident analysis.

Based on this, the probability or consequences of any accident previously evaluated is not increased by the proposed changes.

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

Single control rods can be withdrawn to permit control rod recoupling in OPERATIONAL MODES 3 and 4 under existing TS. The proposed change will merely expand this allowance to other control rod maintenance and testing activities performed in OPERATIONAL MODES 3 and 4. The revision to Specification 3/4.10.A provides additional assurance that the one-rod-out interlock is OPERABLE in OPERATIONAL MODES 3 and 4.

The additional control rod maintenance and testing activities which could be performed in OPERATIONAL MODES 3 and 4 are permitted by the existing TS in OPERATIONAL MODES 1, 2 and 5. Examples of activities which could be performed include venting of control rods following a reactor scram or control rod drive system outage, normal control rod insertion/withdrawal timing and adjustment, control rod scram time testing and control rod friction testing.

Based on this, the proposed changes do not create the possibility of a new or different kind of accident from those previously evaluated.

Specification 3/4.10.A is revised to ensure the one-rod-out interlock is OPERABLE, enhancing the assurance that the plant will prevent the withdrawal of more than one control rod in the manner currently assumed. Expanding the applicability of this existing requirement to OPERATIONAL MODES 3 and 4 similarly does not create the possibility of a new or different kind of accident from those previously evaluated.

3) Involve a significant reduction in the margin of safety because:

The TS currently permit single control rod withdrawal for the purpose of control rod recoupling when in OPERATIONAL MODES 3 or 4 if the one-rod-out interlock is OPERABLE. This change merely allows additional activities for which a single control rod may be withdrawn in OPERATIONAL MODES 3 or 4, with the same restriction that the one-rod-out interlock is OPERABLE.

While the TS currently allow limited control rod withdrawal in OPERATIONAL MODES 3 and 4 provided the one-rod-out interlock is OPERABLE, no explicit surveillance requirements for the one-rod-out interlock exist while in OPERATIONAL MODES 3 or 4. The proposed changes to the Applicability statement in TS 3/4.10.A will result in applicability of the Surveillance Requirements for the one-rod-out interlock whenever control rod withdrawal is performed in OPERATIONAL MODES 3 and 4.

Together, the OPERABILITY requirements for the one-rod-out interlock and the SDM requirements of TS 3.3.A will continue to ensure that the reactor will be maintained subcritical during single control rod withdrawals. Therefore, this change will not involve a significant reduction in the margin of safety.

As described, the proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems required to mitigate accident conditions. Neither are new or significantly different modes of operation proposed. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

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 Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: October 2, 1996

Description of amendment request: Relocation of Radiological Effluent Technical Specifications for Units 1 and 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:

Criterion 1 - Does not Involve a Significant Increase in the probability or Consequences of an Accident Previously Evaluated.

The proposed changes are considered administrative in nature. These changes alter only the location of programmatic controls and procedural details relative to radioactive effluents, radiological environmental monitoring, solid radioactive wastes, and associated reporting requirements. Compliance with applicable regulatory requirements will continue to be maintained. In addition, the proposed changes do not alter the conditions and assumptions in any of the Safety Analysis Report (SAR) accident analyses. Since the SAR accident analyses remain bounding, the radiological consequences previously evaluated are not adversely affected by the proposed changes.

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

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

The proposed changes do not involve any changes to the configuration or method of operation any plant equipment. The proposed changes are considered administrative in nature. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure have been identified as a result of the proposed changes. Also, there will be no change in types or increase in the amounts of any radioactive effluents released offsite.

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

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

The proposed changes do not involve nay actual change in the methodology used in the control of radioactive effluents, solid radioactive wastes, or radiological environmental monitoring. These changes are considered administrative in nature and provide for the relocation of procedural details outside the Technical Specifications. This change adds appropriate administrative controls in the Technical Specifications to provide continued assurance of compliance with applicable regulatory requirements.

Therefore, this change does not involve a significant reduction in the margin of safety. I21Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve significant hazards consideration.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: October 2, 1996

Description of amendment request: Relocation of Selected Technical Specifications Instrumentation Requirements Allowed by Generic Letter 95-10

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

Criterion 1 - Does not Involve a Significant Increase in the probability or Consequences of an Accident Previously Evaluated.

The [Nuclear Regulatory Commission] NRC issued Generic Letter (GL) 95-10 to allow licensees to relocate certain instrumentation requirements to licensee controlled documents or programs. The staff has concluded that the specifications listed in the GL were not required to be included in the technical specifications as required by 10 CFR 50.36. The staff concluded that the instrumentation addressed in these specifications are not related to dominant contributors to plant risk.

The specifications included in this amendment request are being relocated to the Technical Requirements Manual (TRM). Once in the TRM, future changes to these requirements will be controlled under 10 CFR 50.59. By controlling future changes under 10 CFR 50.59, NRC review and approval will be requested for changes exceeding the regulatory threshold of an unreviewed safety question.

This amendment request does not remove or modify any of the instrumentation requirements for either unit. This amendment request does not affect any of the accident initiators, conditions or assumptions for any of the accidents previously evaluated. Therefore, this change does not involve a significant increase in the probability of any accident previously evaluated.

This amendment request is administrative in nature and does not affect any system or component functional requirements. This change does not affect the operation of the plant or affect any component that is used to mitigate the consequences of any accident. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

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

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

The relocation of existing requirements from the technical specifications to other licensee controlled documents is considered administrative in nature. This change does not modify or remove any plant instrumentation requirements. This proposed change will not affect any plant system or structure, nor will it affect any system functional or operability requirements. Consequently, no new failure modes are introduced as a result of this change. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment request represents a relocation of a portion of the information previously located in each unit's technical specification instrumentation section to other licensee controlled documents that ate controlled under 10 CFR 50.59. The proposed change is administrative in nature because the instrumentation requirements for the facility remain the same.

The proposed change does not represent a change in the configuration or operation of the plant. Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve significant hazards consideration.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: December 2, 1996

Description of amendment request: The proposed Technical Specification (TS) Change Request will permit the use of 10 CFR Part 50 Appendix J, Option B, Performance-Based Containment Leakage Testing for Type A, B and C leak rate testing. TSs 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 4.6.1.6 and 4.6.1.7 are revised and Section 6.15 is added establishing the Containment Leakage Rate Testing Program. The Bases are revised to reflect this change. Minor editorial changes are included in this request. Waterford Steam Electric Station is planning to have a Containment Leakage Rate Testing Program in place prior to the next scheduled refueling outage. This program will be in accordance with the guidelines contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-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:

The proposed change will not affect the assumptions, design parameters, or results of any accident previously evaluated. The proposed change does not add or modify any existing equipment. The proposed changes will result in increased intervals between containment leakage tests determined through a performance based approach. The intervals between such tests are not related to conditions which cause accidents. The proposed changes do not involve a change to the plant design or operation. Therefore, this change does not involve a significant increase in the probability of any accident previously evaluated.

NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' contributed to the technical bases for Option B of 10 CFR 50 Appendix J. NUREG-1493 contains a detailed evaluation of the expected leakage from containment and the associated consequences. The increased risk due to lengthening of the intervals between containment leakage tests was also evaluated and found acceptable. Using a statistical approach, NUREG-1493 determined the increase in the expected dose to the public from extending the testing frequency is extremely small. It also concluded that a small increase is justifiable due to the benefits which accrue from the interval extension. The primary benefit is in the reduction in occupational exposure. The reduction in the occupational exposure is a real reduction, while the small increase to the public is statistically derived using conservative assumptions. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

The proposed change does not involve modifications to any existing equipment. The proposed change will not affect the operation of the plant or the manner in which the plant is operated. The reduced testing frequency will not affect the testing methodology. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not change the performance methodology of the containment leakage rate testing program. However, the proposed change does affect the frequency of containment leakage rate testing. With an increased frequency between tests, the proposed change does increase the probability that a increase in leakage could go undetected for a longer period of time. Operational experience has demonstrated the leak tightness of the containment buildings has been significantly below the allowable leakage limit.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rates. The limitation on containment leakage rate is designed to ensure the total leakage volume will not exceed the value assumed in our accident analysis. The margin of safety for the offsite dose consequences of postulated accidents directly related to containment leakage is maintained by meeting the 1.0 La acceptance criteria. The proposed change maintains the 1.0 La acceptance criteria. 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

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

Dates of amendment request: December 9, 1996

Description of amendment request: The licensee proposed to modify specifications for selected cycle-specific reactor physics parameters to refer to the St. Lucie Unit 1 Core Operating Limits Report (COLR) for limiting values. Minor administrative changes are also included. The proposed Technical Specification (TS) changes utilized the guidance provided in Generic Letter 88-16 and are intended to be consistent with the Standard Technical Specifications for Combustion Engineering Plants (NUREG-1432, Revision 1).

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

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

The proposed amendment relocates the calculated values of selected cycle-specific reactor physics parameter limits from the TS to the COLR, and includes minor editorial changes which do not alter the intent of stated requirements. The amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Parameter limits specified in the COLR for this amendment are not changed from the values presently required by Technical Specifications. Future changes to the calculated values of such limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process. Assumptions used for accident initiators and/or safety analysis acceptance criteria are not changed by this amendment. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed

amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment relocates the calculated values of cycle specific reactor physics limiting parameters to the COLR and will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design configuration of St. Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed

amendment would not involve a significant reduction in a margin of safety.

The cycle specific parameter limits being relocated to the COLR by this amendment have not been changed from the values presently required by the TS, and a requirement to operate the plant within the bounds of the limits specified in the COLR is retained in the individual specifications. Future changes to the calculated values of these limits by the licensee may only be developed using NRC-approved methodologies, must remain consistent with all plant safety analysis limits addressed in the Final Safety Analysis Report (FSAR), and are further controlled by the 10 CFR 50.59 process. As discussed in Generic Letter 88-16, the administrative controls established for the values of cycle specific parameters using the guidance of that letter assure conformance with 10 CFR 50.36. Safety analysis acceptance criteria are not being altered by this amendment. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, 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: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: M. S. Ross, Attorney, Florida Power & Light, 11770 US Highway 1, North Palm Beach, Florida 33408

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: December 20, 1996

Description of amendment request: The licensee proposed to delete a footnote associated with TS 2.1.1, ''Reactor Core Safety Limits,'' which requires reactor thermal power to be limited to 90% of 2700 Megawatts thermal for Cycle 14 operation beyond 7000 Effective Full Power Hours [EFPH]. The thermal power limit was required pending completion of a Small Break Loss of Coolant Accident (SBLOCA) reanalysis that demonstrated acceptable results using input assumptions corresponding to an increased number of steam generator tubes being plugged. The SBLOCA reanalysis was completed and included with the submittal.

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

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

The proposed change will allow full Cycle 14 operation at 100% of rated power (2700 MWth), by deleting the requirement to

derate to 90% of rated power prior to exceeding 7000 EFPH. This restriction was imposed in the NRC transmittal letter for License Amendment 145 for SBLOCA considerations when considering the increased SGTP [steam generator tube plugging]

level of 30% plus or minus 7%. All Final Safety Analysis Report (FSAR) events, other than SBLOCA were evaluated at 100% of rated thermal power and showed no significant increases in the probability or consequences of accidents previously evaluated.

The SBLOCA was reanalyzed to demonstrate continued compliance with 10 CFR 50.46 criteria. There is no impact of the proposed change on any FSAR accident initiator. The plant configuration and systems remain unchanged.

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

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

This proposed amendment removes the requirement in the Technical Specifications to derate to 90% of 2700 MWth for Cycle 14 operation beyond 7000 EFPH. There will be no change to the modes of operation of the plant. The plant configuration and the design functions of all the safety systems remain unchanged.

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The impact of the proposed change on available margin to the acceptance criteria for Specified Acceptable Fuel Design Limits (SAFDL), primary and secondary over-pressurization, peak containment pressure, potential radioactive releases, 10 CFR 50.46 requirements for the large break LOCA, and existing limiting conditions for operation has been evaluated and addressed in the reduced RCS [reactor coolant system] flow operating license Amendment No. 145. A requirement to derate to 90% of 2700 MWth was imposed based on the SBLOCA analysis. The small break LOCA analysis with 30% plus or minus 7% SGTP

supported operation up to 7000 EFPH at 100% of rated thermal power. A reanalysis of SBLOCA with the limiting end-of-cycle conditions at 100% of rated power, demonstrates continued compliance with 10 CFR 50.46 criteria.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.The NRC staff has reviewed the licensee's analysis and, based on thisreview, 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: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: M. S. Ross, Attorney, Florida Power & Light, 11770 US Highway 1, North Palm Beach, Florida 33408

NRC Project Director: Frederick J. Hebdon

GPU Nuclear Corporation, Docket No. 50-289, Three Mile Island, Unit 1, Dauphine County, Pennsylvania

Date of amendment request: December 3, 1996

Description of amendment request: This amendment will incorporate certain improvements from the Standard Technical Specifications for B&W Plants (NUREG-1430).

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

GPU Nuclear has determined that this Technical Specification Change Request involves no significant hazards consideration as defined in 10 CFR 50.92 because:

1. Operation of the facility in accordance with the proposed amendment would not

involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment deletes limiting condition for operation (LCOs) from the TMI-1 Technical Specifications that are no longer required to be addressed in Technical Specifications per 10 CFR 50.36(c)(2)(ii). The proposed amendment also deletes a Surveillance requirement from the TMI-1 Technical Specifications. This surveillance requirement has no corresponding LCO and is formatted in the typical LCO format. These items are addressed in licensee controlled documents. This proposed amendment incorporates relaxation of selected timeclocks and surveillances frequencies consistent with NUREG 1430 and adds a timeclock to a unique LCO. The proposed changes do not modify the operation, limits or controls of systems, structures or components relied upon to prevent or mitigate the consequences [of] accidents previously evaluated. Also, the reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Therefore, this change does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure modes are created by the proposed changes.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment does not change any operating limits for reactor operation.

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

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

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

NRC Project Director: John F. Stolz

Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment requests: October 25, 1996

Description of amendment requests: The proposed amendments would incorporate the requirements of 10 CFR Part 50, Appendix J, Option B for containment leakage tests. In addition, the amendments would add a new section to Technical Specifications, which establishes the requirements of the containment leakage rate testing program, consistent with the Improved 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:

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

The proposed changes provide a mechanism within the TS for implementing a performance-based leakage rate test program which was promulgated by the revision to 10 CFR Part 50 to incorporate Option B to Appendix J. The proposed changes do not involve any physical or operational changes to structures, systems or components. The current safety analyses and safety design basis for the accident mitigation functions of the containment, the airlocks, and the containment isolation valves are maintained. Since the allowable containment leakage is still maintained within the analyzed limit assumed in the accident analysis, there is no adverse effect on either onsite or offsite dose consequences. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve any physical or operational changes to structures, systems or components. No new failure mechanisms beyond those already considered in the current plant safety analyses are introduced. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Extending containment leakage rate test intervals from those currently provided in the Technical Specifications to those provided for in 10 CFR (Part) 50 Appendix J, Option B may slightly increase the risk due to an increased likelihood of containment leakage corresponding to the increased testing intervals. However, this is somewhat compensated by the corresponding risk reduction benefits received from the reduction in component cycling, stress, and wear associated with the increased intervals. When considering the total integrated risk, which includes all analyzed accident sequences, the possible additional risk associated with increasing test intervals is negligible.

The NRC letter to NEI (Nuclear Energy Institute) dated November 2, 1995, recognizes that changes similar to the proposed changes at PINGP (Prairie Island Nuclear Generating Plant) are required to implement Option B of 10 CFR (Part) 50, Appendix J. In NUREG-1493, ''Performance-Based Containment Leak-Test Program'', dated September 1995, which forms the basis for the Appendix J revision, the NRC concludes that adoption of performance-based testing will not significantly reduce the margin of safety. The containment leak rate data and component performance history at PINGP are consistent with the conclusions reached in NUREG-1493 and NEI 94-01. Thus, the proposed license amendments do not involve a significant reduction in a margin of safety and will continue to support the regulatory goal of ensuring an essentially leak-tight containment boundary.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

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

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

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

NRC Project Director: John N. Hannon

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 amendment request: November 18, 1996

Description of amendment request: The amendments would amend the Technical Specifications for Susquehanna Steam Electric Station (SSES), Units 1 and 2 by increasing the maximum isolation times for the reactor core isolation cooling inboard warm-up line isolation valves (HV129F088 and HV249F088) from 3 seconds to 12 seconds, the high pressure core injection inboard warm-up line isolation valves (HV-155F100 and HV-255F100) from 3 seconds to 6 seconds and the reactor recirculation process sample line (RRPSL) isolation valves (HV143F019 and HV243F019) from 2 seconds to 9 seconds.

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.

Chapters 6,9, and 15 of the FSAR [final safety analysis report], current operating cycles Reload Summary Reports for Units 1 and 2, Design Basis Document DBD046 (Seismic and Hydrodynamic Loads), and NUREG-0776 (Safety Evaluation Report for SSES), were reviewed to determine if the proposed action has an effect on the spectrum of analyzed anticipated operational transients or postulated design basis accidents.

The proposed modifications involve replacing the pilot solenoid valves on the Reactor Recirculation Loop ''B'' Process Sample Line Isolation Valve (HV1/243F019) and the inboard RCIC [reactor core isolation cooling] and HPCI [high pressure core injection] Steam Warm-Up Line Isolation Valves (HV-1/249F088 and HV-1/255F100). They do not alter any system operation or control logic other than to increase the time it takes for the associated containment isolation valve to close. As discussed above, the effects of the increased isolation times for RCIC and HPCI impacted lines are bounded by the larger parallel lines with isolation times much greater than the new isolation times for the smaller lines. In the case of the Reactor Recirculation Loop ''B'' Process Sample Line, the worst case scenario for a line of that size is addressed in FSAR Section 15.6.2 and the results have been found acceptable. In fact, the line breakage event analyzed in the FSAR section postulates a break outside containment that is not isolable and that does not require operator action for up to 10 minutes.

The modifications enhance isolation valve performance by ensuring proper operation in the event of a degraded air system.

Failures within the Process Sampling, RCIC or HPCI systems or their components are not postulated as causes of accident scenarios nor is increasing the stroke time of the subject containment isolation valves [HV-1/243F019]. These systems provide safety features utilized to mitigate the consequences of the accidents. However, the failure mode of the replacement solenoid valve is similar in each case to that of the solenoid valve being replaced in that it closes upon loss of power or loss of air supply. The current ability of the plant design to meet the single failure criterion is unchanged by this modification.

Based on the above discussion, the proposed action does not involve a significant increase in the probability or consequences of an accident as previously evaluated.

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

Chapters 6, 9, and 15 of the FSAR were reviewed to determine if the proposed action [valve replacement with increased isolation times for associated HPCI, RCIC, RRPSL valves] has the potential of creating a postulated initiating event which is different than the analyzed anticipated operational transients or postulated design basis accident addressed. The review did not identify a postulated initiating event which would create the possibility for an accident of a different type due to replacing the pilot solenoid valves of the affected Reactor Recirculation LOOP ''B'' Process Sample Line or RCIC or HPCI Steam Warm-Up Line isolation valves.

Also, the Reactor Recirculation Process Sample Line, as part of the Process Sampling System described in FSAR section 9.3.2.3, does not perform any safety functions. It is simply an alternate means for in line reactor water chemistry monitoring upon the loss of the RWCU system, and its loss does not create any possibility for unevaluated accidents or malfunctions.

Thus, replacing the pilot solenoid valves on the affected Reactor Recirculation Process Sample Line, RCIC Steam Warm-Up Line, and HPCI Steam Warm-Up Line isolation valves as well as relocating the Process Sample Line solenoid valve for EQ [equipment qualification] purposes 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 a margin of safety.

The proposed action involves replacing existing pilot solenoid valves on containment isolation valves for the Process Sampling, RCIC, and HPCI Systems, as listed above, with direct acting solenoid valves to ensure proper valve operation in the event of a degraded air or gas system as well as relocating the Process Sampling pilot solenoid valve for EQ purposes.

a. Reactor Recirculation Loop ''B'' Process Sample Line

The limiting condition for the operation of the Reactor Recirculation Loop ''B'' Process Sample Line Inboard Isolation Valve (HV-1/243F019) is governed by Technical Specification Section 3/4.6.3 and its Bases which presently requires this valve to close within 2 seconds as defined in Technical Specification Table 3.6.3-1. The proposed modifications involve replacing the pilot solenoid valve of the normally open isolation valve (HV-1/243F019) with a direct acting pilot solenoid valve as well as relocating the pilot solenoid valve to assure an EQ life which supports a 24 month operating cycle. The combined effects of a lower flow coefficient and relocating the solenoid valve will require an increase in the Technical Specification Table 3.6.3-1 isolation time from 2 seconds to 9 seconds.

This increase in isolation time does not reduce the margin of safety as defined in the Technical Specification Section Basis, because breakage of lines of this size is addressed in the Susquehanna SES [steam electric station] FSAR Section 15.6.2 and the results found acceptable. In fact, the line breakage event analyzed postulates a break outside containment that is not isolable and that does not require operator action for up to 10 minutes. Also, it is noted that the outboard isolation valve, HV-1/243F020, also closes on the same containment isolation signal, and its Technical Specification isolation time limit remains 2 seconds.

The failure mode of the affected Reactor Recirculation Loop ''B'' Process Sample Line Inboard isolation valve is to close on loss of power or air supply, therefore, the proposed modifications do not affect the operability of the isolation valve or reduce the margin of safety.

b. RCIC

The limiting condition for operation of the RCIC system is governed by Technical Specification Section 3/4.7.3 and its Bases which requires RCIC to be operable as the primary non-ECCS source of emergency core cooling. The proposed modifications involve replacing the pilot solenoid valve of the normally closed Steam Warm-Up Line Isolation Valve (HV-1/249F088). This valve can be manually opened in the absence of an isolation signal to permit steam from the reactor to pressurize and warm the steam supply line downstream of the HV-1/249F007 valve.

Installation of the direct acting solenoid valve will require an increase in the Technical Specification Section 3/4.6.3 isolation time for the RCIC Steam Warm-Up Line Isolation Valve (HV-1/249F088) from 3 seconds to 12 seconds but does not reduce the margin of safety as defined in the Technical Specification Section Basis. The increase in closure time for the HV-1/249F088 isolation valve does not compromise the overall line isolation due to the fact that the impact of these 1'' warm up line valves is enveloped by the impact of the much larger 4'' RCIC inboard and outboard isolation valves (HV-1/249F007 and HV-1/249F008), which remain open an additional 8 seconds before isolating. The 4'' valves are the limiting components for providing containment isolation for this line.

The failure mode of the affected RCIC Steam Warm-Up Line Isolation Valve is to close, if open, on loss of power or air supply, therefore, the proposed modifications do not affect the operability of the isolation valve or reduce the margin of safety.

c. HPCI

The limiting condition for operation of the HPCI system is governed by Technical Specification Section 3/4.5.1 and its Bases which requires HPCI to be operable for proper Emergency Core Cooling System operation. Operability includes the HPCI pump and a flow path capable of taking suction from the suppression pool and delivering the water to the reactor vessel. The proposed modifications involve replacing the pilot solenoid valve of the normally closed Steam Warm-Up Line Isolation Valve (HV-1/255F100). This valve can be manually opened in the absence of an isolation signal, to permit steam from the reactor to pressurize and warm the steam supply line downstream of the HV-1/255F002 valve.

Installation of the direct acting solenoid valve will require an increase in the Technical Specification Section 3/4.6.3 isolation time for the HPCI Steam Warm-Up Line Isolation Valve (HV-1/255F100) from 3 seconds to 6 seconds but does not reduce the margin of safety as defined in the Technical Specification Section Basis. The increase in closure time for the HV-1/255F100 isolation valve does not compromise the overall line isolation due to the fact that the impact of these 1'' warm up line valves is enveloped by the impact of the much larger 10'' HPCI inboard and outboard isolation valves (HV-1/255F002 and HV-1/255F003) which remain open an additional 44 seconds before isolating. The 10'' valves are the limiting components for providing containment isolation for this line.

The failure mode of the affected HPCI Steam Warm-Up Line Isolation Valve is to close, if open, on loss of power or air supply, therefore, the proposed modifications do not affect the operability of the isolation valve or reduce the margin of safety.

Thus, based on a review of the Technical Specification, their Bases, the FSAR and NUREG 0776 (Safety Evaluation Report for SSES), the replacement of the pilot solenoid valves does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket No. 50-388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: December 18, 1996

Description of amendment request: The amendment would change the Susquehanna Steam Electric Station Unit 2 Technical Specifications to reflect the use of a 24-month operating cycle and the use of the ATRIUM-10 fuel design. The amendment includes changes to two definitions in Section 1, inclusion of new minimum critical power ratio safety limits in Sections 2.1.2 and 3.4.1.1.2, changes in Section 5.3.1 to reflect the new fuel design, and the listing of Siemens Power Corporation topical reports in Section 6.9.3.2.

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

The applicable sections of the FSAR [Final Safety Analysis Report] are Chapters 5,6.3,9, and 15 of the FSAR. Chapter 5 discusses the results of the ASME overpressure analyses for the reactor pressure boundary. Chapter 6.3 discusses the LOCA [loss-of-coolant accident]. Chapter 9 discusses fuel storage and handling. Chapter 15 describes the transient and accident analyses, a majority of which have been generically dispositioned to be non-limiting. A discussion of the impact of the Technical Specification changes is provided below.

The change to Definitions 1.2 and 1.3 makes the definitions applicable to ATRIUM-10. There are no effects on safety functions from this change.

A cycle specific MCPR [minimum critical power ratio] Safety Limit analysis was performed for PP&L [Pennsylvania Power & Light Company] by SPC [Siemens Power Corporation]. This analysis used NRC approved methods described in Technical Specification Reference 13 (ANF-524(P)(A), Revision 2 and Supplement 1 Revision 2.). The SAFETY LIMIT MCPR calculation statistically combines uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and the uncertainty in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The resulting two-loop and single-loop values (Technical Specification sections 2.1.2 and 3.4.1.1.2) are included in the proposed change. Thus, the cladding integrity and its ability to contain fission products is not adversely affected.

The change to the Design Features (Section 5.3) increases the allowable enrichment. Analyses have demonstrated that the ATRIUM-10 fuel will remain subcritical (k-effective<0.95) in both the spent fuel pool and the new fuel vault. Thus, the change to allowable enrichment has no impact on safety functions. The description of a fuel assembly (Section 5.3) is also revised to reflect the ATRIUM-10 central water channel, and reference to an active fuel length of 150 inches was deleted. This change reflects the physical characteristics of the ATRIUM-10 fuel and has no impact on the probability or consequences of an event.

Included in the revised Technical Specifications via reference (Section 6.9.3.2) are additional NRC approved methodology reports. The NRC approved topical reports contain methodology which is used to assure safe operation of Unit 2 with ATRIUM-10 fuel. These methodologies assure that the core meets appropriate margins of safety for all expected plant operational conditions ranging from refueling and cold shutdown of the reactor through power operation. Thus, the results obtained from the analyses will provide assurance that the reactor will perform its design safety function during normal operation and design basis events.

The BASES changes for Section 2.1.1 (THERMAL POWER,Low Pressure or Low Flow) reflect that the Safety Limit is valid for both 9x9-2 and ATRIUM-10.

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

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

The changes to the Unit 2 Technical Specifications (Definitions, MCPR safety limits, Design Features, and inclusion of methodology references) to allow use of ATRIUM-10 fuel do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. The referenced methodology added to Section 6.9.3.2 contains NRC approved acceptance criteria. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility or a new or different kind of accident from any accident previously evaluated.

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

The applicable Technical Specification Sections include 1.0, 2.0, 3/4.4, 5.3, and 6.9.3.2.

The changes to the Unit 2 Technical Specifications discussed in Item 1 above (Definitions, MCPR Safety Limits, Design Features, and inclusion of methodology references) to allow use of ATRIUM-10 fuel do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The NRC approved methods detailed in the references added to Section 6.9.3.2 maintain an equivalent margin of safety as currently defined in the bases of the applicable Technical Specification sections.

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

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

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

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

NRC Project Director: John F. Stolz

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, BrownsFerry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: December 11, 1996 (TS 386)

Description of amendment request: The proposed amendment changes the as-found tolerance for the main steam system safety/relief valves (S/RV) from plus or minus 1% to plus or minus 3%. The licensee states that the proposed change is consistent with methodology submitted by the Boiling Water Reactor Owners Group (BWROG) and approved by the NRC.

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.

TVA [the Tennessee Valley Authority, the licensee] is proposing a change to the ''as-found'' tolerances for the S/RV set points. This proposed TS [technical specification] amendment does not alter the frequency of verifying the S/RV lift set points, or the number of S/RVs required to be operable. The amendment does not involve physical changes or modifications to the S/RVs, or change the operating mode or safety function of the S/RVs. The safety lift set points will still be required to be set within a tolerance of plus or minus 1% following testing.

S/RV actuation is not a precursor to any design basis accident analyzed in the BFN [Browns Ferry Nuclear Plant] UFSAR [Updated Final Safety Analysis Report]. Therefore, this change does not increase the probability of any previously evaluated accident.

Generic considerations related to the set point tolerances were addressed in NEDC-31753P [BWROG In-Service Pressure Relief Valve Technical Specification Licensing Topical Report] and previously reviewed by NRC. In accordance with the NRC SER [Safety Evaluation Report, see letter from A. C. Thadani, NRC to C. L. Tully, BWROG, dated March 8, 1993] on utilizing the NEDC results, certain plant specific evaluations were performed to support the proposed change. Specifically, the current Unit 2 reload licensing report includes the transient analyses for the anticipated operational occurrences and the limiting overpressurization transient utilizing the plus or minus

3% S/RV set point tolerance and were performed in accordance with NRC approved methods. The alternate operating modes were also included in the reload licensing report. These analyses concluded there is adequate margin to design core thermal limits and pressure limits for the reactor vessel. The corresponding Unit 3 core reload licensing report for the next operating cycle (starts in March 1997) is in progress and will also use the plus or minus 3% S/RV set point tolerance. Prior to the return of Unit 1 to service, the same reload analysis will be performed. Similar results to those for Unit 2 are expected.

The operation of high pressure injection systems have been determined not to be adversely affected by the proposed change. LOCA [loss of coolant accident] response, containment hydrodynamic loads, pump and valve performance, and instrumentation performance were likewise satisfactorily evaluated. Therefore, this proposed change does not significantly increase the consequences of any previously evaluated accident.

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

The proposed change does not involve a modification to plant equipment. No new failure modes are introduced. Plant systems will continue to function and no new system interactions are introduced 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.

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

The proposed change has been analyzed in accordance with NRC approved methodology and the margins of safety for the design basis accidents and transients analyzed in Chapter 14 of the BFN UFSAR have not been significantly reduced. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: December 11, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 1.0, ''Definitions,'' TS Table 1.2, ''Frequency Notation,'' TS Section 3/4.3, ''Instrumentation,'' and TS Section 3/4.5, ''Emergency Core Cooling Systems.'' Surveillance requirements would be modified to account for the increase in the fuel cycle, consistent with Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,'' dated April 2, 1991. Administrative changes consistent with the fuel cycle change 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:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the subject Technical Specifications (TS) 3.4.3.1.1, Reactor Protection System Instrumentation; TS 3/4.3.2.1, Safety Features Actuation System Instrumentation; TS 3/4.3.2.2, Steam and Feedwater Rupture Control System Instrumentation; TS 3/4.3.3.1, Radiation Monitoring Instrumentation; TS 3/4.3.3.5.2, Remote Shutdown Instrumentation; TS 3/4.3.3.6, Post-Accident Monitoring Instrumentation, TS 3/4.5.1, Emergency Core Cooling Systems (ECCS), Core Flooding Tanks; and TS 3/4.5.2, Emergency Core Cooling Systems, ECCS Subsystems - T_{avg} greater than or equal to 280°F. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

Review results of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because little, if any, potential for an increase in a failure rate of a system or component was identified during these reviews.

These proposed revisions are consistent with NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991.

The proposed revision to Technical Specification Table 1.2, Frequency Notation, and the related proposed revision from an

''R'' frequency notation to an ''E'' frequency notation for Technical Specification Surveillance Requirements that are remaining on an 18 month frequency, are administrative in nature, do not change current actual Technical Specification requirements, and do not affect previously evaluated accidents.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

The proposed revision to Technical Specification Table 1.2, Frequency Notation, and the related proposed revision from an ''R'' frequency notation to an ''E'' frequency notation for Technical Specification Surveillance Requirements that are remaining on an 18 month frequency, are administrative in nature, do not change current actual Technical Specification requirements, and do not affect previously evaluated accidents.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

Review results of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because little, if any, potential for an increase in a failure rate of a system or component was identified during these reviews. No changes are being proposed to the type of testing being performed, only to the length of the surveillance test interval.

The proposed revision to Technical Specification Table 1.2, Frequency Notation, and the related proposed revision from an ''R'' frequency notation to an ''E'' frequency notation for Technical Specification Surveillance Requirements that are remaining on an 18 month frequency, are administrative in nature, do not change current actual Technical Specification requirements, and do not affect the manner in which systems and components are being operated or tested.

3. Not involve a significant reduction in a margin of safety because the review results of the historical 18 month surveillance data and maintenance records identified little, if any, potential for an increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy is not being changed by these proposed changes.

The proposed revision to Technical Specification Table 1.2, Frequency Notation, and the related proposed revision from an ''R'' frequency notation to an ''E'' frequency notation for Technical Specification Surveillance Requirements that are remaining on an 18 month frequency, are administrative in nature, do not change current actual Technical Specification requirements, and do not reduce the margin of safety.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, therefore there are no significant reductions 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 Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

NRC Project Director: Gail H. Marcus

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.

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

Date of application for amendment: May 1, 1996, as supplementedNovember 26, 1996.

Brief description of amendment: The proposed amendment will modify Table 3.1.1, ''Reactor Protection System (SCRAM) Instrumentation Requirement,'' Table 3.2.C.1, ''Instrumentation That Initiates Rod Blacks,'' and Technical Specification 3/4.4, ''Standby Liquid Control.''

Date of issuance: December 27, 1996

Effective date: December 27, 1996

Amendment No.: 169

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

Date of initial notice in Federal Register: June 6, 1995 (61 FR 28606) The November 26, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 27, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 26, 1996, as supplemented on September 3, 1996, September 18, 1996, two submittals dated October 14, 1996, October 22, 1996, two submittals dated November 8, 1996, and December 17, 1996.

Brief description of amendments: The amendments allow Commonwealth Edison Company to control the reactor coolant system pressure and temperature limits for heatup, cooldown, low temperature operation and hydrostatic testing. They also revise the reactor vessel material surveillance program specimen withdrawal schedule such that the Unit 2 removal of capsule X is delayed until 19 Effective Full Power Years.

Date of issuance: December 20, 1996

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 177 and 164

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 25, 1996 (61 FR 50341). The September 3, 1996, September 18, 1996, two submittals dated October 14, 1996, October 22, 1996, two November 8, 1996, and December 17, 1996, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 25, 1996 (NRC-96-0003)

Brief description of amendment: The amendment revises the testing requirements used to determine the operability of the charcoal in the engineered safety feature systems.

Date of issuance: December 23, 1996

Effective date: December 23, 1996, with full implementation within 45 days

Amendment No.: 110

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

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40014) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 11, 1996, as supplemented December 17, 19, and 26, 1996

Brief description of amendments: The amendments approve changes to the Updated Final Analysis Report (UFSAR), and require that the changes be submitted with the next update of the UFSAR pursuant to 10 CFR 50.71(e). The associated Safety Evaluation delineates the staff's review and findings regarding the one-time emergency power engineered safeguards functional test.

Date of issuance: January 2, 1997

Effective date: January 2, 1997

Amendment Nos.: 220, 220, 217

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the Updated Final Safety Analysis Report. Public comments requested as to proposed no significant hazards consideration: Yes. (61 FR 66699 December 18, 1996) The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by January 2, 1997, as corrected to read January 17, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The December 17, 19, and 26, 1996, letters provided additional information that did not change the scope of the December 11, 1996, application and initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated January 2, 1997.

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

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

Date of amendment request: May 31, 1996

Brief description of amendment: The amendment revises the technical specifications to increase the amount of trisodium phosphate (TSP) dodecahydrate located in the containment sump storage baskets.

Date of issuance: December 30, 1996

Effective date: December 30, 1996

Amendment No.: 179

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

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40025) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 1, 1996

Brief description of amendment: This amendment revised the Technical Specifications 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.

Date of issuance: December 30, 1996

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

Amendment No.: 80

Facility Operating License No. NPF-85. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 6, 1996 (61 FR 57490) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1996.No significant hazards consideration comments received: No

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

Notice Of Issuance Of Amendment To Facility Operating License And FinalNo Significant Hazards Consideration Determination

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

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

Date of amendment request: December 6, 1996

Brief description of amendment request: The amendment would revise Technical Specification (TS) Section 2.1 and its associated TS Basis to reflect the change in the Minimum Critical Power Ratio Safety Limit due to the use of GE13 fuel product line and the cycle-specific analysis performed by General Electric Company (GE), for Limerick Generating Station, Unit 2, Cycle 5.

Date of publication of individual notice in Federal Register: December 23, 1996 (61 FR 67582)

Expiration date of individual notice: January 22, 1997

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

Dated at Rockville, Maryland, this 8th day of January 1997.

For the Nuclear Regulatory Commission

Jack W. Roe,

Director, Division of Reactor Projects - III/IV, Office of Nuclear Reactor Regulation

[Doc. 97–848 Filed 1–14–97; 8:45 am]

BILLING CODE 7590–01–F

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Reclearance of Information Collection, OPM Form 805 Series

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that OPM will submit a request to the Office of Management and Budget for reclearance of the OPM Form 805 Series that collects information from the public. OPM Form 805, Application to be Listed Under the Voting Rights Act of 1965, is used to elicit information from persons applying for voter registration under the authority of the Voting Rights Act of 1965. The requirements for voter eligibility vary from State to State; therefore, OPM Form 805 is a blanket number covering a number of forms which conform to the individual State's requirements. For a number of years, there have been forms for 10 States: Alabama, Arizona, Georgia, Louisiana, Mississippi, New Mexico, North Carolina, South Carolina, Texas (English and Spanish language versions), and Utah. Because OPM has never been asked to list voters in Arizona, New Mexico, North Carolina, and Utah, the approval of these four forms is being permitted to lapse at the request of the Voting Rights Section in the Civil Rights Division of the Department of Justice. The form requires 20 minutes to complete. Approximately 10 individuals complete the form annually for a total public burden of 4 hours. For copies of this proposal call James M. Farron on (202) 418–3208 or e-mail to jmfarron@opm.gov.

DATES: Comments on this proposal should be received on or before March 17, 1997.

ADDRESSES: Send or deliver comments to—Steven R. Cohen, Assistant Director for Merit Systems Oversight, Office of Personnel Management, 1900 E Street, NW., Room 7677, Washington, DC 20415–0001.

FOR FURTHER INFORMATION CONTACT: P. Kaziah Clayton on (202) 606–2531 or e-mail to pkclayto@opm.gov.

U.S. Office of Personnel Management.

Lorraine A. Green,

Deputy Director.

[FR Doc. 97–993 Filed 1–14–97; 8:45 am]

BILLING CODE 6325–01–M

PHYSICIAN PAYMENT REVIEW COMMISSION

Sunshine Act Meeting

AGENCY: Physician Payment Review Commission.

ACTION: Notice of meeting.

SUMMARY: The Commission will hold its next public meeting on Thursday, January 23, 1997, and Friday, January 24, 1997, at the Washington Marriott, 1221 22nd Street NW., Washington, DC, in the DuPont Salon. The meetings are tentatively scheduled to begin at 9:00 a.m. each day. In preparation for its March 31 report, the Commission expects to discuss such issues as vulnerable populations, academic health centers, quality of care, and federal premium contributions. It will also review draft chapters on PSOs, access in Medicare managed care, Medicare PPOs, risk adjustment, secondary insurance for Medicare beneficiaries, consumer protections in managed care, and Medicare Fee Schedule issues. Final agendas will be mailed on January 17, 1997 and will be available on the Commission's web site (WWW.PPRC.GOV) at that time.

ADDRESS: 2120 L Street NW., Suite 200, Washington, DC 20037. The telephone number is 202–653–7220.

FOR FURTHER INFORMATION CONTACT: Annette Hennessey, Executive Assistant, at 202–653–7220.

SUPPLEMENTARY INFORMATION: If you are not on the Commission mailing list and wish to receive an agenda, please call 202–653–7220 after January 16, 1997.

Lauren LeRoy, Ph.D.,

Executive Director.

[FR Doc. 97–1129 Filed 1–13–97; 3:08 pm]

BILLING CODE 6820–SE–M

UNITED STATES POSTAL SERVICE BOARD OF GOVERNORS

Sunshine Act Meeting; Notification of Items Added to Meeting Agenda

DATE OF MEETING: January 6, 1997.

STATUS: Closed.

PREVIOUS ANNOUNCEMENTS: 61 FR 65092, December 10, 1996; and 61 FR 68081, December 26, 1996.

CHANGE: At its meeting on January 6, 1997, the Board of Governors the United States Postal Service voted unanimously to add two items to the agenda of its closed meeting held on that date:

4. Consideration of Personnel and Compensation Issues.
5. Changes to the FY 1997 Advertising Budget.

CONTACT PERSON FOR MORE INFORMATION: Thomas J. Koerber, Secretary of the Board, U.S. Postal Service, 475 L'Enfant Plaza, S.W., Washington, D.C. 20260–1000. Telephone (202) 268–4800.

Thomas J. Koerber,

Secretary.

Certified to be a true copy of the original document.

Neva R. Watson,

Alternate Certifying Officer.

[FR Doc. 97–1057 Filed 1–10–97; 4:41 pm]

BILLING CODE 7710–12–M

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Existing collection in use without an OMB Number: