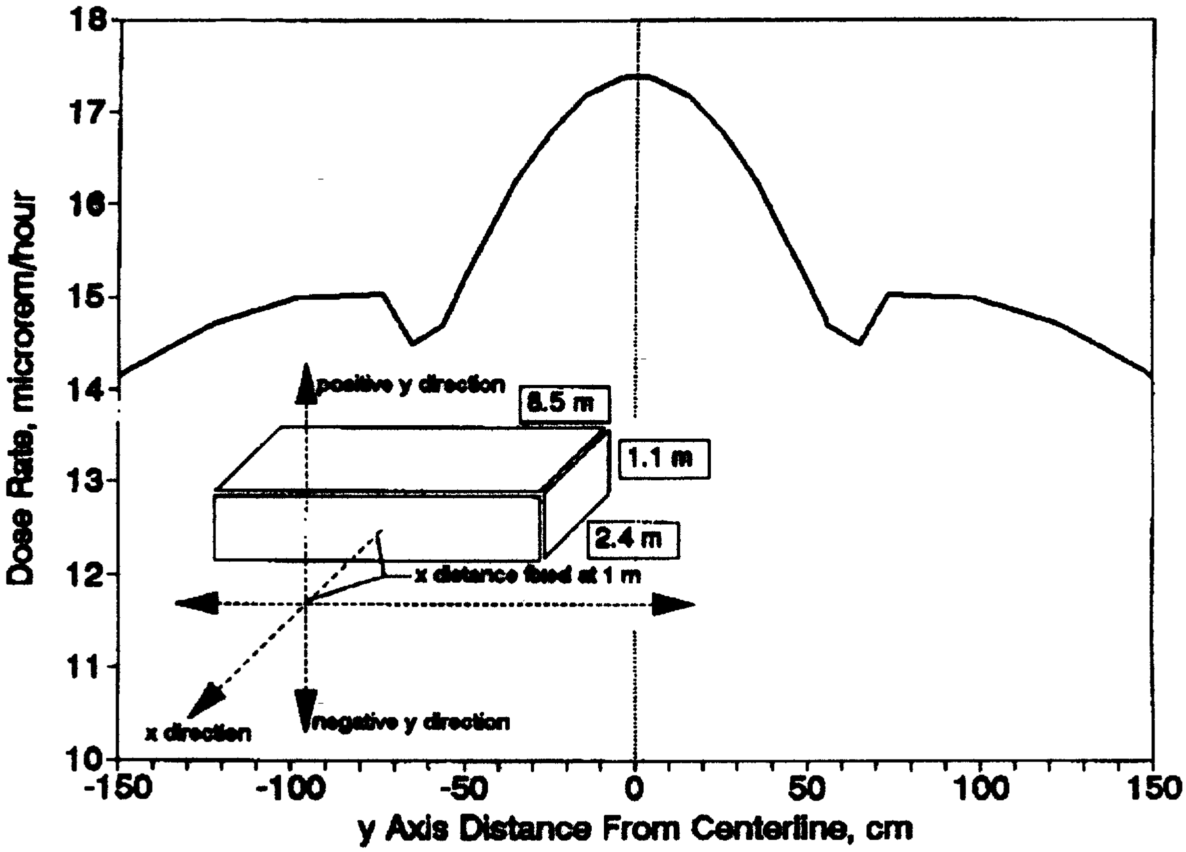

Figure 8. Dose Rates as a Function of Vertical Distance Offset from Center Point of Trailer-Type Container

BILLING CODE 7590–01–P

Dated at Rockville, Maryland, this 11th day of January, 1996.

For the Nuclear Regulatory Commission.

Michael F. Weber,

Chief, Low-Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 96–703 Filed 1–19–96; 8:45 am]

BILLING CODE 7590–01–O

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 21, 1995, through January 4, 1996. The last biweekly notice was published on January 3, 1996 (61 FR 174).

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

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

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

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

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

By February 21, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendments request: December 19, 1995

Description of amendments request: The proposed amendments would allow the implementation of the recently approved Option B to 10 CFR Part 50, Appendix J. This new rule allows for a performance-based option for determining the test frequency for containment leakage rate testing. The proposed amendment would modify Technical Specifications (TS) 1.7, 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, and 3/4.6.3 and the Bases of TS 3/.6.1.2. It would also create a new TS 6.16.

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

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

The proposed Technical Specification (TS) changes will result in generally increased intervals between containment leakage rate tests determined through a performance based approach. The interval between such tests are not related in any way to conditions which cause accidents. Plant structures, systems, and components will not be operated in a different manner as a result of the proposed TS change, therefore, the proposed changes will not increase the probability of an accident previously evaluated.

Containment leakage may result from accidents which are evaluated in the Updated Final Safety Analysis Report. The proposed TS changes may result in a small, but acceptable, increase in post-accident containment leakage. This increase is calculated as a statistical expectation using the probability that leakage through a penetration will exceed the administrative limit and through the increased time needed to detect such excess leakage. NUREG–1493, which is the technical basis for 10 CFR Part 50, Appendix J, Option B, contains a detailed evaluation of the expected leakage and its consequences.

The increased risk due to the lengthening of the intervals between Type A, B, and C leakage rate tests is also evaluated in NUREG–1493. Using a statistical approach, NUREG–1493 determined that the increase in expected dose to the public, resulting from extending the testing interval, is extremely small. NUREG–1493 concluded that the small increase is justifiable due to the benefits which accrue from interval extension. The primary benefit is the reduction in occupational exposure. The reduction, on a per person basis, is orders of magnitude greater than the marginal, potential increase in dose to the public. The reduction in occupational exposure is a real reduction, while the small increase in dose to the public is statistically derived using conservative assumptions. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change only incorporates the performance based approach authorized in the new Option B to Appendix J of 10 CFR Part 50. The interval extensions allowed, through this approach, do not have the potential for creating the possibility of new or different kinds of accidents from those previously evaluated. Plant structures, systems, and components will not be operated in a different manner as a result of the TS change and, therefore, will not introduce any new or different failure modes or initiators.

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

The proposed Technical Specification does not alter the allowable containment leakage rate. The proposed change replaces the current, prescriptive testing requirements with a new performance based approach for establishing the testing intervals therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: December 21, 1995.

Description of amendment request: The proposed amendment would revise the Calvert Cliffs Nuclear Power Plant, Unit No. 1, Technical Specifications (TSs). The requested change would allow the use of cladding materials other than Zircaloy or ZIRLO. A Temporary Exemption was issued on November 28, 1995 (60 FR 62483) approving the loading of four (4) lead fuel assemblies (LFAs) into the Unit No. 1 reactor vessel during cycles 13, 14, and 15. The technical basis for the Exemption, which is the same basis for the requested TS amendment, was provided in the Baltimore Gas and Electric Company (BGE) submittal dated July 13, 1995. The submittal addressed the safety significance of operating with 4 LFAs in Calvert Cliffs Nuclear Power Plant, Unit No. 1, reactor vessel during cycles 13, 14, and 15.

Specifically, BGE proposes to add a statement to TS 5.2.1, ''Fuel Assemblies,'' indicating, for Cycles 13, 14, and 15 only, advanced cladding material may be used in 4 lead test assemblies as described in a approved Temporary Exemption dated November 28, 1995.

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

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

The proposed change is to add an approved temporary exemption to the Unit 1 Technical Specifications allowing the installation of four lead fuel assemblies. These four assemblies use an advanced cladding material which is not specifically permitted by existing regulations or Calvert Cliffs' Technical Specifications. A temporary exemption to allow the installation of these assemblies was approved on November 28, 1995. The addition of this approved temporary exemption to Technical Specification 5.2.1 is simply intended to allow their installation under the provisions of the temporary exemption. The license amendment is effective only as long as the exemption is effective. The addition of the approved temporary exemption to Unit 1 Technical Specification 5.2.1 does not change the probability or consequences of an accident previously evaluated.

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

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

The proposed Technical Specification change adds an approved temporary exemption to Technical Specification 5.2.1 for Unit 1. This change does not add any new equipment, modify any interfaces with existing equipment, change the equipment's function, or change the method of operating the equipment. The proposed change does not affect normal plant operations or configuration. Since the proposed change does not change the design, configuration, or operation, it could not become an accident initiator.

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

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

The proposed change is to add an approved temporary exemption to the Unit 1 Technical Specifications allowing the installation of four lead fuel assemblies. These four assemblies use an advanced cladding material which is not specifically permitted by existing regulations or Calvert Cliffs' Technical Specifications. A temporary exemption to allow the installation of these assemblies was approved on November 28, 1995. The addition of this approved temporary exemption to Technical Specification 5.2.1 is simply intended to allow their installation under the provisions of the temporary exemption. The license amendment is effective only as long as the exemption is effective. This amendment does not change the margin of safety by adding a reference to an approved, temporary exemption to the Technical Specifications.

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

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: December 7, 1995.

Description of amendment request: The proposed amendments will remove the Technical Specification (TS) requirements for the main feedwater pump discharge pressure switch input to the Anticipatory Reactor Trip System (ARTS) and the Emergency Feedwater System (EFDW).

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:

No. The accidents addressed within the Oconee Final Safety Analysis Report (FSAR) have been reviewed with respect to this proposed Technical Specification amendment request. The probability or consequences of any accident previously evaluated is not significantly increased by the proposed amendment. Emergency Feedwater is required for the mitigation of some accidents and the availability of this system will be unaffected by this proposed revision. Both manual and automatic actuation of the EFDW system on a loss of main feedwater will remain.

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

No. This amendment eliminates a portion of the automatic actuation circuitry for EFDW and ARTS. This circuitry removal does not create the possibility of a new or different kind of accident as the design of the circuitry is to sense a loss of main feedwater and supply a signal for the initiation of ARTS and EFDW. A loss of main feedwater signal will continue to be supplied to ARTS and EFDW; however, this loss will be sensed by low hydraulic oil pressure on the Main Feedwater Pumps (ARTS and EFDW) and low steam generator level (EFDW only) rather than by a low Main Feedwater Pump discharge pressure. Since a loss of Main Feedwater will continue to be recognized, the system will continue to function as before. Hence, no new or different accidents will be created.

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

No. The margin of safety will not be significantly reduced as an actuation signal to ARTS and EFDW will continue to be generated by a loss of Main Feedwater. Consequently, ARTS and EFDW will continue to perform the safety function required for accident mitigation.

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

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

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: November 22, 1995.

Description of amendment request: The proposed amendments will upgrade existing TS [Technical Specification] 3/4.4.6.1 for the Reactor Coolant System Leakage Detection Instrumentation by adapting the Standard Technical Specifications for Combustion Engineering Plants (NUREG–1432), Specification 3.4.15, to both St. Lucie units. The proposal is consistent with the NRC Final Policy Statement on Technical Specifications Improvements (58 FR 39132).

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 Reactor Coolant System (RCS) Leakage Detection Instrumentation Systems are not accident initiators, and their operational status is not a consideration in determining the probability of occurrence of accidents previously evaluated. The proposed revision to the related Limiting Condition for Operation (LCO) 3/4.4.6.1 does not involve a change to the configuration or method of operation of any equipment that is used to mitigate the consequences of an accident, nor do the changes alter any assumptions made involving initial plant conditions in the safety analyses. 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.

The proposed revision to LCO 3/4.4.6.1 is administrative in nature and will not result in a change to the physical plant or the modes of plant operation defined in the Facility License. The revision does not involve the addition or modification of equipment nor does it alter the design of 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 RCS Leakage Detection Systems are designed to provide diverse methods to assist in the detection and location of unidentified leakage that may be associated with potential pressure boundary degradation. These systems provide no equipment control or accident mitigation functions, and are not associated with the safety margin established for protection from analyzed Loss of Coolant Accidents. The proposed revision to LCO 3/4.4.6.1 does not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin; and simply adapts the corresponding and previously reviewed specification from the Standard Technical Specifications for Combustion Engineering Plants, NUREG–1432, to the St. Lucie units. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussions and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendment involves no significant hazards consideration.

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

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

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews, Director.

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

Date of amendment request: December 5, 1995.

Description of amendment request: The proposed amendment revises the submittal date in the Annual Exposure Data Report which brings Oyster Creek into conformance with 10 CFR 20.2206 and relaxes an overly restrictive administrative requirement.

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 changes do not:

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

This change is administrative in nature and has no effect on the operation of the plant. This change will not increase the probability or consequence of an accident previously evaluated.

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

Operation of the facility in accordance with this proposed change will not create the possibility for an accident or malfunction of a different type from any accident previously evaluated. The proposed amendment does not modify any system (component) operation or maintenance activity. The facility will continue to be operated within the limits of existing accident analysis and margins of safety.

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

This change brings the submittal date for the Annual Exposure Data Report into conformance with 10 CFR 20.2206 and relaxes an overly restrictive administrative requirement. Since the proposed change does not alter any system hardware or design basis, the margin of safety is not reduced.

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

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

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

NRC Project Director: Phillip F. McKee.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa.

Date of amendment request: November 15, 1995.

Description of amendment request: The proposed amendment would revise the requirements for the End of Cycle Recirculation Pump Trip logic to match more closely the assumptions applicable to the turbine trip events for which it was installed. The surveillance requirements are also proposed to be revised, based on those same assumptions.

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

1. The proposed Technical Specification (TS) amendment will not significantly increase the probability or consequences of any previously evaluated accidents. The [End of Cycle] (EOC) [recirculation pump trip] RPT system was installed to preclude

violation of reactor fuel limits, and the system will be preserved for that purpose. In the event that system is not available, an operating penalty will be imposed on the [Minimum Critical Power Ratio] MCPR limit to assure sufficient margin to the limit to preclude fuel damage during the postulated turbine trip events.

The change to the ''Minimum Operable Channels per Trip System'' will assure that inputs monitoring both the turbine control valve fast closure and the turbine stop valve closure will be available to initiate (EOC)RPT.

The change to the ''Applicable Operating Mode'' is an editorial change which reflects the existing hardware bypass.

The change to Action 81 in TS Table 3.2–G will assure that when the (EOC)RPT system does not meet the minimum TS availability requirements, the [safety limit minimum critical power ratio] SLMCPR will not be challenged. By imposing an [operating limit minimum core power ratio] OLMCPR penalty for continued operation, the fuel thermal limits will not be challenged, since the (EOC)RPT system was installed to accomplish the same goal. No increase in the consequences of the turbine trip events will result from this change. The OLMCPR penalty is dependent on cycle-specific parameters and will therefore be included in the cycle-specific [Core Operating Limits Report] COLR.

The change to the surveillance interval results in (EOC)RPT logic channel functional tests being performed once per quarter instead of once per month. The change also revises the allowed out-of-service time (AOT) for testing from two hours to six hours. These changes are consistent with the Improved Standard Technical Specifications, NUREG–1433, Revision 1. The (EOC)RPT is initiated by instruments common to the Reactor Protection System (RPS) (i.e., turbine stop valve closure and turbine control valve fast closure). The surveillance interval and AOT changes for these instruments were evaluated in ''Technical Specification Improvement Analysis for BWR Reactor Protection System,'' NEDC–30851P–A, March 1988, for the RPS function. Although the (EOC)RPT functions were not explicitly identified in that document, these changes can be considered bounded by that analysis. The basis for this conclusion is similar to the basis established for the control rod block instrumentation common to the RPS, as documented in ''Technical Specification Improvement analysis for BWR Control Rod Block Instrumentation,'' NEDC–30851P–A, Supplement 1, October 1988. Failure of the (EOC)RPT function could potentially lead to exceeding the SLMCPR, similar to the consequences of an unmitigated rod withdrawal error. The slight increase in risk of a SLMCPR violation due to extending (EOC)RPT surveillance interval and AOT is offset by the same benefits associated with the similar approved surveillance interval and AOT for the RPS. Both the above referenced reports have been approved for application at the DAEC via TS Amendment 193, dated April 14, 1993.

The changes to the ''Operating Modes for which Surveillance Required'' are clarifications and will result in a more efficient utilization of resources. By stating that the surveillance applies only when the (EOC)RPT system is OPERABLE, the surveillances will not be performed needlessly. During the early part of an OPERATING cycle, the (EOC)RPT is not required to mitigate a turbine trip, and therefore, may be bypassed. At the time when the (EOC)RPT is assumed to be OPERABLE pursuant to the analysis, it will be made OPERABLE unless accepting the penalty on the OLMCPR is preferable. The result of the proposed change will still be that the (EOC)RPT is demonstrated OPERABLE at any time when it is required.

The change to the acceptance criteria for response time testing reflects a recent review of the analytical assumptions and the testing methodology. The (EOC)RPT is assumed to interrupt power to the recirculation pump motor within 175 milliseconds after initiation of either turbine stop valve closure or turbine control valve fast closure. The response time test only measures a portion of the complete trip (the rest was measured as part of start-up testing). The portion measured is dependent on which trip input is being tested. The turbine control valve closure is sensed by a pressure switch monitoring the hydraulic fluid controlling the valve and therefore has no delay between valve motion and initiation of the (EOC)RPT logic. The turbine stop valve closure is sensed by position switch. Since this switch is set to initiate (EOC)RPT at 10% valve closed, there is a brief delay between the beginning of valve motion and initiation of the (EOC)RPT logic. The respective proposed response time tests account for these differences, as described in the footnotes on TS page 3.2–36, and demonstrate that the measured portions of the action are within allowed time periods.

None of the proposed changes will significantly increase the probability of any accident previously evaluated because the (EOC)RPT is not an initiator of any of those events. None of the proposed changes will significantly increase the consequences of an accident because the (EOC)RPT system serves to prevent a turbine trip event from exceeding the fuel SLMCPR, and it will continue to perform in that capacity at any time when it is required to assure margin to the SLMCPR.

2. The proposed changes will not add a new or different kind of accident because the plant will not be operated in a different way. By allowing the implementation of a penalty on OLMCPR in lieu of reducing reactor power, the risk of a plant transient is reduced. Similarly, the surveillance interval and AOT extensions will also result in fewer plant power reductions for testing.

The (EOC)RPT initiates a trip of the recirculation pumps and any TS change affecting that system cannot result in an effect on any system other than those pumps. Consequently, no new accidents are postulated as a result of this proposed change.

3. The proposed change will not result in a significant reduction in any margin of safety. The (EOC)RPT performs to assure adequate margin to the SLMCPR. The proposed change will preserve that function and require that additional margin to the SLMCPR be imposed for those times when the (EOC)RPT is not OPERABLE. The other changes are proposed because they assure correct (EOC)RPT function (inputs and response times).

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: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis, & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: December 14, 1995.

Description of amendment request: The proposed amendment would modify Technical Specification 3.4.2, ''Flow Control Valves (FCVs),'' by deleting the requirement to verify that the average rate of movement of each reactor recirculation system FCV is limited to less than or equal to 11% per second in the opening and closing directions (Surveillance Requirement 3.4.2.2).

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 Clinton Power Station (CPS) Updated Safety Analysis Report (USAR) evaluates three specific events related to operation of the reactor recirculation flow control valves (FCVs). The impact of the proposed change on each of these events is discussed below.

The loss of coolant accident (LOCA) analysis described in USAR Section 6.3.3.7.2 assumes that the FCVs fail ''as is'' in the event of a LOCA. This feature is assured by electronic interlocks in the FCV control circuitry and periodically verified as required by Technical Specification (TS) Surveillance Requirement (SR) 3.4.2.1. The design of these interlocks and the testing requirements are not affected by this proposed change.

The Recirculation Flow Controller Failure—Decreasing Flow transient analyses are described in USAR Section 15.3.2, and the Recirculation Flow Controller Failure—Increasing Flow transient analyses are described in USAR Section 15.4.5. Since the control circuitry for the FCVs has been modified such that the capability to operate in a master controller mode has been eliminated, each FCV is now individually controlled, and the possibility that a single failure could affect operation of more than one FCV has also been eliminated. As a result, fact closure and fast opening of both FCVs are no longer postulated for CPS. Thus, the surveillance (SR 3.4.2.2) associated with verifying that FCV movement is within the assumptions of the analyses for fast closure and fast opening of both FCVs can be deleted.

With respect to fast closure and fast opening of individual FCVs, the modification performed during the fifth refueling outage only affected the electronic master control of the FCVs and did not affect the hydraulic limitations of the FCVs. Conservative analyses, component testing, and the Initial Startup Test program provide confidence that individual FCV stroke rates assumed in the transient analyses will not be exceeded over the life of the plant. These analyses and conditions are sufficient to assure individual FCV stroke rates are adequately limited without the periodic performance of a specific test.

In addition to the above, the modification did not add any new failure modes to the design of the individual FCV controllers. In fact, failure modes associated with misoperation of the common master controller have been eliminated from the control circuit design. The modification did not alter any of the features associated with initiators of any LOCA or features which assure that the FCVs fail ''as is'' in the event of a LOCA.

Based on the above, Illinois Power (IP) has concluded that this request does not increase the probability or the consequences of any accident (or transient) previously evaluated.

(2) USAR Sections 15.3.2 and 15.4.5 describe the plant response to malfunctions of FCV control failures, and USAR Section 6.3.3.7.2 describes the assumptions made with respect to FCV failures and their impact on the LOCA analysis. The proposed change (and the associated modification prompting the proposed change) does not affect any other structures, systems, or components beyond the FCVs. All associated failure modes thus remain within the scope of the failure modes previously considered. As a result, IP has concluded that the proposed change cannot create the possibility of an accident not previously evaluated.

(3) This request does not involve any change to the requirements or design associated with initiation or mitigation of a LOCA. The consequences of transients associated with fast closure and fast opening of reactor recirculation system FCVs are bounded by the consequences of other transient events and thus are not utilized in establishing plant operating limits. Although the control circuitry for the FCVs was modified during the fifth refueling outage, that modification did not affect the hydraulic failure modes of the FCVs. Further, the modification did not add any new failure modes to the design of the individual FCV controllers. In fact, failure modes associated with misoperation of the common master controller have been eliminated from the control circuit design. As a result, assumed FCV operation during analyzed accidents and transients has not been altered. Conservative analysis, component testing, and the Initial Startup Testing program have confirmed that the FCV velocity assumed in the transient analyses will not be exceeded over the life of the plant. Thus, verification of rate of FCV movement in the opening and closing directions need not be performed by periodic testing and SR 3.4.2.2 can be deleted without resulting in a reduction in a margin of safety.

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

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

Attorney for licensee: Sheldon Zabel, Esq., Schiff, Hardin and Waite, 7200 Sears Tower, 233 Wacker Drive, Chicago, Illinois 60606.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: December 14, 1995.

Description of amendment request: The proposed amendment would consist of several changes to the instrumentation sections of the Clinton Power Station Technical Specifications. The proposed changes are required due to engineering reanalyses or plant modifications. The affected instrumentation includes: (1) steam line flow high channels for the Reactor Core Isolation Cooling (RCIC) System, (2) ambient temperature channels in the Residual Heat Removal (RHR) System heat exchanger rooms, (3) reactor vessel pressure channels that provide a permissive for operation of the shutdown cooling mode of the RHR system, and (4) RCIC storage tank water level instrument channels.

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) None of the proposed changes involve a significant increase in the probability or consequences of any accident previously evaluated.

The changes to Table 3.3.6.1–1 Functions 3.a and 3.i are administrative in nature and bring the technical specifications (TS) into conformance with the Clinton Power Station (CPS) as-built design. The reactor core isolation cooling (RCIC) system steam line flow trip Function names have been changed to reflect the elimination of the residual heat removal (RHR) steam condensing mode. However, these trips have not been physically altered and thus will continue to operate as before. As a result of the elimination of the RHR steam condensing mode, the possibility of a leak in the RCIC steam supply resulting in an increase in the RHR heat exchanger room ambient temperature has also been eliminated. Accordingly, the RHR ambient temperature isolation trip is changed to only isolate the RHR system when the RHR heat exchanger room ambient temperature setpoint is exceeded. The Shutdown Cooling System Reactor Vessel Pressure—High function is provided to isolate the shutdown cooling portion of the RHR system since this piping is designed for pressures lower than rated reactor vessel pressure. This interlock (RHR cut in permissive) is provided only for equipment protection to prevent an intersystem LOCA scenario and credit for the interlock is not assumed in the accident or transient analysis in the Updated Safety Analysis Report (USAR).

The proposed change to the setpoint (Allowable Value) is conservative with respect to considerations for shutting the RHR shutdown cooling motor-operated valves and providing overpressurization protection for the low pressure RHR shutdown cooling system piping. With respect to the RCIC storage tank water level setpoints, no accident or transient analysis takes credit for the volume of water in the RCIC storage tank. In addition, the setpoint (Allowable Value) has been changed to ensure RCIC system operation is not adversely affected by a low level in the storage tank.

The proposed changes do not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. In addition, the proposed changes do not affect the ability of the associated instrumentation to operate as assumed in the safety analyses. As a result, the proposed changes will not result in a significant increase in the consequences of any accident previously evaluated.

(2) None of the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes for RHR/RCIC Steam Line Flow—High [are] administrative in nature and will simply make this item description accurate. The RCIC steam supply line no longer supplies any steam to the RHR heat exchanger room. As a result, the associated isolation of the RCIC system is no longer required. The Shutdown Cooling System Reactor Vessel Pressure - High function will still perform as designed. The RCIC Storage Tank Level - Low trip will continue to perform in accordance with design. None of the above listed changes will introduce any new failure modes or changes in plant operation.

As a result, the proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) None of the proposed changes involve a significant reduction in a margin to safety.

The proposed changes for RHR/RCIC Steam Line Flow—High do not involve a significant reduction in a margin of safety because the change is administrative in nature and will simply make the descriptions accurate and consistent with completed modifications. The elimination of RCIC system isolation in response to a high RHR room ambient temperature is no longer required due to the elimination of the RHR steam condensing mode. Removing the RHR room ambient temperature isolation of the RCIC will reduce the number of unnecessary isolations of RCIC. The Shutdown Cooling System Reactor Vessel Pressure - High function will still perform as designed. The proposed change to the setpoint (Allowable Value) is conservative with respect to considerations for shutting the RHR shutdown cooling motor-operated valves and providing overpressurization protection for the low pressure RHR shutdown cooling system piping. The Allowable Value for the RCIC Storage Tank Level - Low Function has been changed to be more conservative to ensure the RCIC and HPCS systems will perform their system safety function. No credit is taken for the volume in the RCIC storage tank for the HPCS or RCIC systems in performing their safety-related functions.

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

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

Attorney for licensee: Sheldon Zabel, Esq., Schiff, Hardin and Waite, 7200 Sears Tower, 233 Wacker Drive, Chicago, Illinois 60606

NRC Project Director: Gail H. Marcus.

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

Date of amendment requests: December 19, 1995 [AEP:NRC:1215B]

Description of amendment requests: The proposed amendments would modify the technical specifications to replace the existing scheduling requirements for overall integrated and local containment leakage rate testing with a requirement to perform the testing in accordance with 10 CFR Part 50, Appendix J, Option B. Option B allows test scheduling to be adjusted based on past performance.

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

This amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes to the T/Ss do not affect the assumptions, parameters, or results of any UFSAR [updated final safety analysis report] accident analysis. The proposed changes do not change the acceptance criteria for containment leakage limits and do not modify the response of the containment during a design basis accident. The proposed amendment does not add or modify any existing equipment. The proposed Types A, B, and C testing schedules will be consistent with Appendix J Option B to 10 CFR 50 which was developed based on analytical efforts documented in NUREG–1493 [Performance-Based Containment Leak-Test Program]. The analysis confirms previous observations of insensitivity of population risks from severe reactor accidents to containment leakage rates. Based on these considerations, it is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed changes do not involve physical changes to the plant or changes in plant operating configuration. The proposed changes only remove the restrictive schedular requirements for conducting Types A, B, and C testing from the T/Ss and substitute the schedule specified in Appendix J Option B to 10 CFR 50 and Regulatory Guide 1.163 [Performance-Based Containment Leak-Test Program]. Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

Based on NUREG–1493, Regulatory Guide 1.163, and the rule posting in the Federal Register (60 FR 49495), the margin for safety presently provided is not significantly reduced by the proposed change to a performance-based test interval for Types A, B, and C tests. Although the changes allow more flexibility in scheduling tests, the proposed amendment continues to ensure reactor containment system reliability by periodic testing in full compliance with 10 CFR 50, Appendix J Option B. Based on these considerations, it is concluded that the changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: August 15, 1995, as supplemented November 14, 1995.

Description of amendment request: The proposed amendment would modify the Monticello Technical Specifications (TS) to: (1) revise the main steam line isolation valve leak rate test acceptance criterion to be based upon the combined maximum flow path leakage for all four main steam lines of 46 standard cubic feet per hour (scfh) in lieu of the current limit of 11.5 scfh per valve; (2) revise the operability test interval for the drywell spray header and nozzles from 5 years to 10 years; and (3) revise TS 3/4.7.a.2, Primary Containment Integrity, to remove information specific to the primary containment leakage rate testing program and replace it with a commitment to abide by the requirements of 10 CFR Part 50, Appendix J, Option B, Section III.A, for Type A testing.

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

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

The proposed amendment is limited to changes to the surveillance testing requirements applicable to the main steam line isolation valves [MSIVs] allowable leakage criteria, drywell nozzles test interval, and method of applying Appendix J test requirements. With respect to monitoring main steam [line] isolation valve performance, the proposed criteria are equivalent to the current criteria ensuring that leakage past the valves would be within acceptable limits under accident conditions. These surveillance tests are performed while the plant is in a cold shutdown condition at a time when the equipment is not required to be operable. Performance of the tests themselves are not input or consideration in any accident previously evaluated, thus the proposed change will not increase the probability of any such accident occurring.

The proposed amendment will not adversely affect the function, operation, or reliability of the equipment, nor will it diminish the capability of the equipment to perform as required during an accident.

Combining the maximum per valve leak rate into an overall maximum leakage limit does not increase the overall permissible leakage and thus has no significant impact on the consequences of previously analyzed accidents since the combined leak rate of the main steam line isolation valves, and thus the contribution of the valves to overall primary containment leakage as used for analysis purposes, is unchanged. Extending the drywell nozzle test interval has been shown by industry experience to not compromise safety, and removing the specifics of primary containment leakage testing from the Technical Specifications and referencing 10 CFR Part 50 Appendix J does not alter either how actual testing is accomplished nor the acceptance criteria. It has been shown that adopting longer test intervals based on performance, maintains the safety objective for containment integrity while at the same time reducing the burden on licensees, and provides a greater level of worker safety than that provided by the previous rule.

Therefore, there will be no increase in post accident off-site or on-site radiation dose as a result of this amendment. The proposed amendment requires compliance with the regulatory requirements of 10 CFR Part 50, Appendix J Option B, Section III.A, for Type A testing that has previously been reviewed by the NRC and found to be acceptable. Therefore, the amendment will not increase the consequences of any accident previously evaluated.

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

The proposed amendment does not involve any modification to plant equipment or operating procedures, nor will it introduce any new equipment failure modes that have not been previously considered. The proposed amendment is limited to changes in surveillance test frequencies of tests performed while the plant is in cold shutdown when the associated equipment is not required to be operable. We therefore conclude the proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Combining the allowable leak rate for the MSIV's from a per valve limit to an overall limit does not change the total allowable leakage and therefore post accident dose levels remain unchanged. Extending the drywell nozzle surveillance test interval from 5 to 10 years has been shown by industry experience to be acceptable. Extending the intervals between containment integrated leakage tests as authorized by 10 CFR Part 50, Appendix J, Option B, does not change the acceptance criteria nor how testing is accomplished.

Based on these considerations, we conclude the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment requests: December 19, 1995.

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

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

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

The proposed changes simplify the Technical Specifications (TS), meet regulatory requirements for relocated TS, and implement the recommendations of: (1) the NRC's letter dated October 25, 1993, from William T. Russell to the chairpersons of the industry owners groups; (2) the Commissions's Final Policy Statement on TS Improvements; and (3) the recently revised 10 CFR 50.36. Future changes to these requirements will be controlled by 10 CFR 50.54 and 10 CFR 50.59. Any changes that reduce the effectiveness of the Quality Assurance Program will be approved by the NRC prior to implementation. The proposed changes are administrative in nature and do not involve any modifications to any plant equipment or affect plant operation.

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

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

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety-related system performs its function.

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

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

The proposed changes do not alter the basic regulatory requirements and do not affect any safety analyses. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: September 15, 1995.

Description of amendment request: The licensee proposes to extend the surveillance test intervals for the auxiliary electrical systems to support 24-month operating cycles.

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

Operation of the James A. Fitzpatrick plant in accordance with the proposed

Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

The proposed changes increase the interval between auxiliary electrical system functional tests and also propose additional requirements for battery performance testing. These changes are consistent with the guidance provided in Generic Letter 91–04. These changes do not involve any special changes to the plant, nor do they alter the way the auxiliary electrical system functions. Past equipment performance indicates that the test acceptance criteria has been consistently met, providing additional assurance that the longer surveillance interval will not degrade system performance. The proposed changes revise Bases section 4.9 to clarify battery testing requirements and indicate consistence with the length of the 24 month operating cycle. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes increase the interval between auxiliary electrical system functional tests and also propose additional requirements for battery performance testing. These changes are consistent with the guidance provided in Generic Letter 91–04. The proposed changes do not change the ability of the auxiliary electrical systems to provide electrical power during a design basis accident. Past equipment performance indicates that the test acceptance criteria has been consistently met, providing additional assurance performance. The proposed changes do not modify the design or operation of plant equipment, therefore, no new or different failure modes are introduced. The proposed changes revise Basis section 4.9 to clarify battery testing requirements and indicate consistency with the length of the 24 month operating cycle. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes increase the interval between auxiliary electrical system functional tests and also propose additional requirements for battery performance testing. These changes are consistent with the guidance provided in Generic Letter 91–09. The proposed changes do not alter the configuration of the auxiliary electrical system nor change the manner in which the system functions. Operation of the facility remains unchanged by the proposed changes. An evaluation of past equipment performance indicates that auxiliary electrical system operability is not time dependent. The proposed changes revise Bases section 4.9 clarify battery testing requirements and indicate consistency with the length of the 24 month operating cycle. Therefore, a longer surveillance test interval for the station batteries and LPCI [low-pressure coolant injection] batteries will not degrade performance of the auxiliary electrical system and will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: October 25, 1995.

Description of amendment request: The licensee proposes to extend the surveillance test intervals for the containment systems to support 24-month operating cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 40.19(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

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

The proposed changes do not involve any physical changes to the plant, do not alter the way the containment systems function, and will not degrade the performance of the containment systems. The type of testing and the corrective actions required if the subject surveillance fail remains the same. The proposed changes do not adversely affect the availability of the containment systems or affect the ability of the systems to meet their design objectives. A historical review of surveillance test results indicated that there was no evidence of any failures which would invalidate the above conclusions.

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

The proposed changes do not modify the design or operation of the plant and therefore no new failure modes are introduced. No changes are proposed to the type and method of testing performed, only to the length of the surveillance interval. Past equipment performance and on-line testing indicate that longer test intervals will not degrade the containment systems. A historical review of surveillance test results indicated that there was no evidence of any failure which would invalidate the above conclusions.

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

Although the proposed changes will result in an increase in the interval between surveillance tests, the impact on system reliability is minimal. This is based on more frequent on-line testing and the redundant design of the containment systems. A review of past surveillance history has shown no evidence of failure which would significantly impact the reliability of the containment systems. Operation of the plant remains unchanged by the proposed containment system surveillance test interval extensions. The assumptions in the Plant Licensing Basis are not impacted. Therefore the proposed changes do not result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: November 30, 1995.

Description of amendment request: The licensee proposes to extend the surveillance test intervals for the standby liquid control (SLC) system to support 24 month operating cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.19(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

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

The proposed changes do not involve any physical changes to the plant, do not alter any SLC system functions, and will not degrade the performance of the SLC system. The type of testing and the corrective actions required if the subject SLC surveillances fail remain the same. The proposed changes do not adversely affect the availability of the SLC system or the ability of the system to bring the reactor from full power to a cold shutdown condition in the unlikely event that control rods cannot be inserted. A historical review of SLC surveillance test results indicated that there was no evidence of any failures that would invalidate the above conclusions.

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

The proposed changes do not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. No changes are proposed to the type and method of testing performed, only to the length of the surveillance interval. Past equipment performance and on-line testing indicate the longer test intervals will not degrade SLC equipment. A historical review of surveillance test results indicated that there was no evidence of any failures that would invalidate the above conclusions.

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

Although the proposed changes will result in an increase in the interval between surveillance tests, the impact on system reliability is minimal. This is based on more frequent on-line testing of major system components and the redundant design of the SLC system. A review of past SLC surveillance history has shown no evidence of failures that would significantly impact the reliability of the SLC system. The longer testing intervals do not significantly impact the SLC safety margins for SLC normal operation, operation with inoperable components, or sodium pentaborate solution as described in the bases of the Technical Specifications. Operation of the plant remains unchanged by the proposed SLC surveillance interval extensions. The assumptions in the Plant Licensing Basis are not impacted. Therefore, the proposed changes do not result in a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: December 14, 1995.

Description of amendment request: The licensee proposes to incorporate the inservice testing (IST) requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The proposed change adds a new surveillance requirement, 4.0.E, which refers to the requirements of Section XI of the ASME Code and Addenda established by 10 CFR 50.55a(f). Ancillary changes are also required since the proposed specification 4.0.E replaces the surveillance testing requirements of safety related pump and motor-operated valves and extends the surveillance testing frequency of other components from once every month, to coincide with the ASME Code Section XI requirements.

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

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

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

The changes identified in this proposed amendment revise surveillance testing for various systems based upon the Section XI of the American Society of Mechanical Engineers [***] Boiler and Pressure Vessel [***] Code [ASME Code]. None of these changes involves a hardware modification to the plant, a change to system operation, a change to the manner in which the system is used, or a change in the ability of the system to perform its intended function.

The use of Section XI of the ASME [***] Code as a basis for establishing surveillance testing and acceptance criteria will not alter existing accident analyses. This has been acknowledged and accepted by the NRC in the Standard Technical Specifications. The change to surveillance testing frequencies reduces testing at power, increases the availability of systems important to the mitigation of a DBA [design-basis accident], and minimizes component degradation due to excessive testing. The ASME [***] Code, Section XI testing tracks component performance allowing identification of component degradation and the code specifies that if a pump parameter enters the alert range, then the testing frequency is doubled until the cause of the degradation is determined and the condition corrected. Similarly, if a valve stroke time degrades, the valve testing frequency is increased to once per month until the cause is determined and the condition corrected.

The editorial changes are strictly non technical in nature with no effect on existing analyses. They clarify the Technical Specifications by improving the legibility of this document.

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

The proposed changes involve no hardware changes, no changes to the operation of the systems, and do not change the ability of the systems to perform their intended functions. The use of ASME Section XI as the basis for testing involves the same testing alignments and practices previously used as part of either the IST program or Technical Specification Surveillance Requirements. The editorial changes have no effect on plant practices.

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

There are no hardware modifications, changes to system operations, or effect on the ability of systems to perform their intended function associated with the proposed changes. The proposed changes to reference pump and valve testing to Section XI of the ASME [***] Code and remove individual Surveillance Requirements in the Technical Specifications does not relax any controls or limitations. The resulting reduction in test frequency, while reducing the possibility of detecting a degraded component prior to failure, is offset by the increased availability of systems important to plant safety and an associated reduction in component wear and degradation due to excessive testing. Additionally, the ASME testing program evaluates components for degraded performance and will identify such degradation early. There are no safety margins associated with the editorial corrections.

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

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

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

NRC Project Director: Ledyard B. Marsh.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina.

Date of amendment request: December 8, 1995.

Description of amendment request: The proposed changes add a new surveillance requirement to Technical Specification (TS) Section 4.1.2.2 and deletes TS Sections 3/4.1.2.3 and 3/4.1.2.4 associated with the Borations Systems section. TS Section 3/4.9.3 is being revised to assure only one charging pump is capable of Reactor Coolant System injection in the applicable modes and to add a new surveillance requirement to demonstrate this assurance. TS Section 4.5.2.f is being revised to delete specific Emergency Core Cooling System pump testing acceptance criteria and reference acceptance criteria located in the plant Inservice Testing Program. In addition, the licensee has proposed changes to the bases.

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 probability or consequences of an accident previously evaluated is not significantly increased.

The implementation of the above described TS changes will have no impact on the probability of an accident occurring. The testing of the ECCS pumps at a more appropriate point on their characteristic curve is not a precursor to an accident. There is no hardware, software, or testing methodology change proposed that would decrease confidence in the reliability of these systems/components.

The proposed revision to the ECCS Pump testing surveillance will allow greater flexibility for testing and will provide more useful information about the performance capabilities of those pumps.

The deletion of the Reactivity Control System Specifications (Charging Pumps - Operating and Charging Pumps - Shutdown) will have no impact on the capability of the Charging/SI pumps to perform their design function. The additional Action Statement and Surveillance for low temperature overpressure (LTOP) assure that safety analyses remain valid and initial conditions are not changed. The additional Surveillance Requirement for Boration Systems assures that one charging pump will be operable during Modes 5 and 6.

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

This proposed TS change does not involve any changes to station hardware, software, or operating practices. The changes do provide for a revision to the testing methodology used in demonstrating the capability of the ECCS pumps.

This methodology will test the ECCS pumps at a point on the pump's characteristic curve that will more reliably indicate the pump's continued operability at or near the parameters the pump would be required to provide during a postulated accident.

The deletion of the Reactivity Control System Specifications (Charging Pumps - Operating and Charging Pump - Shutdown) will not provide additional challenges to the capability of the plant to meet normal operational needs or mitigate the conditions of a design basis accident. The ECCS Subsystems TS provide similar surveillance requirements to insure continued operability of the Charging/SI pumps. The LTOP TS will now provide requirements to assure that design assumptions are not challenged and RCS integrity is maintained.

Therefore, as the above described change has no impact on plant performance, the possibility of a new or different kind of accident being created as a result of this change is negligible.

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

The change in testing philosophy for ECCS pumps should bring an increase in margin of safety, since testing will be conducted at reference flow points closer to actual pump parameters for accident conditions. For the Residual Heat Removal Pumps this will be conducted quarterly and for the centrifugal charging pumps, they will be tested quarterly on minimum flow and each refueling outage at substantial flow per the Inservice Testing Program.

The surveillance requirements of TS 3/4.1.2.3 and TS 3/4.1.2.4 are essentially the same as those in 3./4.5.2 and 3/4.5.3 (ECCS Subsystems), and the deletion of these requirements will have no adverse impact on margin on safety. The addition of the Action Statement and Surveillance Requirements to 3/4.4.9.3 (Overpressure Protective Systems) provide additional requirements to supplement those above to assure RCS integrity is maintained for all operational modes. The addition of the Surveillance Requirement to 3/4.1.2.1 will provide assurance that reactivity control can be maintained for Modes 5 and 6 through the charging system flow path.

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: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendments request: December 19, 1995.

Description of amendments request: The proposed amendments would replace the requirements associated with the Control Room Emergency Ventilation System with requirements related to the operation of the Control Room Emergency Filtration/Pressurization System and Control Room Air Conditioning System. These changes are technically consistent with the requirements of NUREG–1431, Revision 1, ''Westinghouse Standard Technical Specifications,'' issued on April 7, 1995. Also, a one-time extension to the allowable outage time for the control room recirculation filtration system is included to facilitate implementation of design modifications to enhance the reliability of the control room air conditioning system during the spring of 1996.

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:

Based on the preceding evaluation, the following conclusions are provided with respect to the criteria contained in 10 CFR 50.92.

(1) The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The proposed changes have no impact on the probability of an accident. The control room ventilation systems are support systems which have a role in the detection and mitigation of accidents but do not contribute to the initiation of any accident previously evaluated. Reorganizing the technical specifications by functions have no impact on the course of any accidents previously evaluated. The other changes which are being made improve the ability to mitigate fuel handling accidents. Specifying an allowed outage time (AOT) of 30 days for the cooling of recirculated air while one train is inoperable is based on the significance of the cooling function but does represent an increase in the allowed outage time and thus an increase in the probability that the functions could be unavailable. This increase is not considered significant based on several factors including: the design is based on the worst postulated meteorological conditions; generally, less than design cooling is required and a partial failure in the system may have no impact; and unavailability failure does not create an immediate irreversible impact (i.e., temperature will increase slowly over a period of time); the system could be restored or its loss mitigated without any impact on the course or whatever accident is being considered; and the extended AOT would allow more opportunity to perform major required maintenance and thus may provide an overall improvement in equipment reliability.

In addition, the one-time change to the AOT for the recirculation filtration will not significantly increase the probability or consequences of an accident due to the low probability of an event result[ing] in an airborne release of radioactivity. Such an event requires multiple failures of safety systems that are governed by technical specifications not affected by these changes. In addition, compensatory measures have been identified that limit the potential exposure of control room operators in response to a postulated release.

The net effect of these changes is not significant and, as a result, the changes do not involve a significant increase in the consequences of an accident previously evaluated.

(2) The proposed changes to the Technical Specifications do not increase the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new limiting single failure or accident scenarios have been created or identified due to the proposed changes. Safety-related systems are expected to perform as designed. Although the changes could have a minor impact on the air conditioning system availability, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed changes do not involve a significant reduction in the margin of safety. The changes proposed do not alter the environmental conditions which are to be maintained in the control room during normal operations and following an accident. As a result, the margin of safety for these functions remains the same. Although there is a potential impact on the air conditioning system's postulated availability, there is no impact on the accident analyses. Further, although the one-time AOT extension for the recirculation filtration system increases the system unavailability during the planned CRACS [Control Room Air Conditioning System] design changes, the net effect is a benefit to plant safety due to the enhancement to control room cooling capability. Thus, even if system availability issues were considered an aspect of margin of safety, 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: December 8, 1995 (TS 364).

Description of amendment request: The licensee proposes revision of Units 1, 2, and 3 Technical Specifications (TS) Section 4.7.A to implement the revision to 10 CFR 50, Appendix J. The new rule (Option B) provides a voluntary performance-based testing option for containment leak rate testing. Option B containment leak rate testing requirements are based on system and component performance in lieu of compliance with the current prescriptive requirements.

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

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

The proposed amendment to TS Section 4.7.A is in accordance with Option B to 10 CFR 50, Appendix J. The proposed amendment adds a voluntary performance based option for containment leak rate testing. The changes being proposed do not affect the precursor for any accident or transient analyzed in Chapter 14 of the BFN [Browns Ferry Nuclear Plant] Updated Final Safety Analysis Report (UFSAR). The proposed change does not increase the total allowable primary containment leakage rate. The proposed change does not reflect a revision to the physical design and/or operation of the plant. Therefore, operation of the facility in accordance with the proposed change does not affect the probability or consequences of an accident previously evaluated.

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

The proposed amendment to TS Section 4.7.A is in accordance with the new performance-based option (Option B) to 10 CFR 50, Appendix J. The changes being proposed will not change the physical plant or the modes of operation defined in the facility license. The proposed changes do not increase the total allowable primary containment leakage rate. The changes do not involve the addition or modification of equipment, nor do they alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to TS Section 4.7.A is in accordance with the new option to 10 CFR 50, Appendix J. The proposed option is formulated to adopt performance-based approaches. This option removes the current prescriptive details from the TS. The proposed changes do not affect plant safety analyses or change the physical design or operation of the plant. The proposed change does not increase the total allowable primary containment leakage rate. Therefore, operation of the facility in accordance with the proposed change does not involve a significant reduction in the margin of safety.

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

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 12, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.6.1.1, Containment Systems—Primary Containment—Containment Integrity; TS 3/4.6.1.2, Containment Systems—Containment Leakage; TS 3/4.6.1.6, Containment Systems—Containment Vessel Structural Integrity; TS 3/4.6.5.3, Containment Systems—Shield Building Structural Integrity; and associated Bases. The proposed revisions adopt the provisions of Appendix J, Option B for Type A containment leakage testing as modified by approved exemptions and in accordance with the guidance of Regulatory Guide 1.163. The licensee proposes to delete surveillance requirement (SR) 4.6.1.2, SR 4.6.1.2.b, SR 4.6.1.2.c, and SR 4.6.1.2.i since these requirements contain details that are now included in standards that are referenced by Regulatory Guide 1.163. TS 3/4.6.1.6 and TS 3/4.6.5.3 which address containment building and shield building structural integrity are proposed to be deleted since the requirements are addressed in revised TS 3.6.1.2.a. The licensee proposes to delete the exemption included in Bases

3/4.6.1.2 since it is no longer applicable. Additionally, the licensee proposes to modify the Action statement associated with TS 3.6.1.2 to reflect the action to take if the as-left rather than the as-found leakage exceeds 0.75 L_a.

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 the changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because accident initiators, conditions, or assumptions are not affected by the proposed changes.

The proposed changes to the Technical Specifications implement 10 CFR 50 Appendix J Option B for Type A testing, including visual examinations of the containment vessel and shield building, and make various administrative changes to the Technical Specifications and associated Technical Specification Bases. Therefore, as stated above, these proposed changes do not affect accident initiators, conditions, or assumptions.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, containment isolation, or allowable releases.

The proposed changes involve containment leakage testing and test frequency. The allowable containment leakage rates presently specified in the Technical Specifications remain unchanged.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety, for the reasons cited below.

The proposed changes involve containment leakage testing and test frequency. The allowable containment leakage rates presently specified in the Technical Specifications remain unchanged. The Technical Specifications, under the proposed changes, will continue to ensure containment system reliability by periodic testing performed in full compliance with 10 CFR 50 Appendix J.

As stated in the Federal Register publication of the final rule, 60 FR 49495 dated September 26, 1995, the final rule improves the focus of the regulations by eliminating prescriptive requirements that are marginal to safety. Further, the final rule allows test intervals to be based on system and component performance and provides licensees greater flexibility for cost-effective implementation methods of regulatory safety objectives. The final rule publication also discusses the following specific findings documented in NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' September, 1995, which justify the proposed change in frequency of Type A Integrated Leak Rate Testing (ILRT):

1. The fraction of leakages detected only by ILRT's is small, on the order of a few percent.

2. Reducing the frequency of ILRT testing from 3 every 10 years to one every 10 years leads to a marginal increase in risk.

3. At a frequency of one test every 10 years, industry-wide occupational exposure would be reduced by 0.087 person-sievert (8.7 person-rem) per year.

Based on these considerations, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: 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.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Power Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin.

Date of amendment request: December 13, 1995.

Description of amendment request: The proposed amendments will modify Technical Specification (TS) Sections 15.1, ''Definitions,'' 15.2, ''Safety Limits and Limiting Safety System Settings,'' 15.3, ''Limiting Conditions for Operation,'' and 15.6, ''Administrative Controls.'' The proposed changes would modify the TSs to account for the creation and maintenance of a Core Operating Limits Report.

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 this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

The relocation of the cycle-specific parameters from the Point Beach Nuclear Plant (PBNP) Technical Specifications to the Core Operating Limits Report (COLR) has no impact on plant operation or accident analyses. The proposed changes are administrative in nature. The Technical Specifications will continue to require operation within the core operational limits for each cycle reload calculated by the NRC-approved reload design methodologies. The appropriate actions required if limits are exceeded will remain in the Technical Specifications. The reload report presents the results of a cycle-specific evaluation of accidents and transients addressed in the PBNP Final Safety Analysis Report (FSAR). The cycle-specific evaluation demonstrates that changes in the unit's fuel cycle design and corresponding COLR parameters do not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Operation of this facility under the proposed Technical Specifications will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to relocate the cycle-specific parameters from the Technical Specifications to the COLR is administrative in nature. No change to the design, configuration, or method of operation of the plant is made by this change. The cycle-specific parameters will be determined using NRC-approved methodologies. The Technical Specifications will continue to require operation within the core operating limits and appropriate actions will be taken if the limits are exceeded.

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

3. Operation of this facility under the proposed Technical Specifications will not create a significant reduction in a margin of safety.

Existing Technical Specification operability and surveillance requirements are not reduced by the proposed changes to relocate cycle-specific parameters from the Technical Specifications to the COLR. The cycle-specific COLR limits for reloads will continue to be developed based on NRC-approved methodologies, thereby maintaining accepted margins of safety. The Technical Specifications will still require that the core be operated within these limits and specify appropriate actions to be taken if the limits are violated. Each reload undergoes a 10 CFR 50.59 safety review to assure that operating the unit within the cycle-specific limits will not involve a significant reduction in a margin of safety. Therefore, these 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: Joseph P. Mann Library, 1516

Sixteenth Street, Two Rivers, Wisconsin 54241.

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: December 13, 1995.

Description of amendment request: This license amendment request proposes to revise the 125-volt D.C. Sources Technical Specifications (3.8.2.1 and 3.8.2.2) to include provisions for installed spare chargers, which will be added to the plant design during the next refueling outage. The Onsite Power Distribution Technical Specifications 3.8.3.1 and 3.8.3.2 would be revised to indicate that spare chargers may be connected in place of the primary chargers.

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.

These proposed technical specification changes do not alter the plant design bases nor do they involve any hardware changes that significantly increase the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation capabilities. There will be no increase in the consequences of any accident or equipment malfunction.

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

The proposed technical specification changes do not involve any design bases changes nor are there any changes to the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unaffected. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined, nor will there be any effect in those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on DNBR [departure from nucleate boiling ratio] limits, F_Q, F-delta-H, LOCA [loss-of-coolant accident] PCT [peak cladding temperature], peak local power density or any other 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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: December 13, 1995.

Description of amendment request: This change request proposes revising the minimum and maximum flow requirements for the centrifugal charging pumps (CCPs) and safety injection pumps (SIPs) specified in Technical Specification Surveillance Requirement 4.5.2.h. Specifically, the proposed changes would:

(1) Decrease the minimum limits on the sum of the injection line flow rates, excluding the highest flow rate, from 346 gpm to 330 gpm for the CCPs and from 459 gpm to 450 gpm for the SIPs.

(2) Revise the maximum pump flow rate for the SIP from 665 to 670 gpm, but retain the CCPs maximum pump flow rate at its current value of 556 gpm.

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

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

The proposed change will not result in a condition where the material or construction standards applicable prior to the change are altered. The ECCS [emergency core cooling system] system integrity is not affected by this change, and this change will not affect the ability of the ECCS to fulfill its design functions. This change will modify the pump surveillance criteria to prevent pump runout during the test, but will not affect the method of operation of the system and will not alter the testing method for the pumps. This change will slightly alter the acceptance criteria of the test, but the changes have been determined to be enveloped by the ECCS pump flow and balance criteria assumed in the safety analyses described in the USAR [Updated Safety Analysis Report]. This change will not affect the ability of the ECCS to mitigate the consequences of any previously evaluated accident. The proposed change will not alter, degrade or prevent the response of the ECCS to any accident scenarios evaluated in the USAR. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated in the USAR will be increased by this 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 will alter the existing ECCS pump flow test to prevent pump runout during the test by slightly altering the acceptance criteria of the test. However, the proposed changes have been determined to be enveloped by the ECCS pump flow and balance criteria assumed in the safety analyses described in the USAR. This change will not create a new type of accident or malfunction, and the method and manner of plant operation remains unchanged. This change will not alter the safety functions of the ECCS. The safety design bases in the USAR have not been altered, and no new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this change. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

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

There are no changes being made to any safety limits or safety system settings that would adversely impact plant safety. This proposed change will have no affect on the availability, operability or performance of any safety-related system or component. The analysis results and conclusions of the accidents presented in the current USAR would not be adversely affected by the revised surveillance requirements for the ECCS. This conclusion is drawn based on the evaluation that confirms that the actual ECCS flow characteristics remain consistent with assumptions used in the WCGS [Wolf Creek Generating Station] accident analyses. Specifically, the accident analyses which are limiting with minimized ECCS flow have already been analyzed using revised ECCS flows that were developed based on a more conservative minimum flow than the proposed minimum ECCS flow requirement. For the analyses which are limiting with a higher ECCS flow, the evaluation indicated that a higher pump runout limit proposed for the SIPs would have insignificant effect on the results and conclusions of the analyses. The evaluation also indicated that the ECCS pump operability would not be a concern as a result of increasing the SIPs runout limit because the available runout margin is sufficient to accommodate the cumulative effect of the ECCS performance issues. Based on these reasons, it is concluded that implementation of the proposed changes will have no adverse impact on the ECCS subsystems' operability and their intended safety function. Therefore, the proposed change would not result in a reduction in a margin of safety.

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

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of amendment request: December 13, 1995.

Description of amendment request: This license amendment request proposes revising Surveillance Requirement 4.1.3.1.3 to delete the requirement for performing the control rod drop surveillance test with T_{avg} greater than or equal to 551°F. This would allow performing this test with T_{avg} below 551°F. This change will also add justification for performing the rod drop test with T_{avg} below 551°F to Bases Section 3/4.1.3, ''Movable Control Assemblies.''

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

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

The proposed change will not result in a condition where the material or construction standards applicable prior to the change are altered. The rod control system integrity is not affected by this change, and this change will not affect the ability of the system to fulfill its design function. This change will allow the control rod drop test to be performed at lower temperatures than currently allowed, but will not affect the method of operation of the system and will not alter the drop time criterion of the test. This change will not affect any fission product barrier, and will not affect the integrity of any fuel assembly or the reactor internals. Thus this change will not affect the ability of the rod control system to mitigate the consequences of any previously evaluated accident. The proposed change will not alter, degrade or prevent the response of the rod control system to any accident scenarios evaluated in the USAR [Updated Safety Analysis Report]. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated in the USAR will be increased by this 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 will alter the existing rod drop test to allow the test to be performed over a range of temperatures, but will not alter the rod drop time criterion of the test. This change will not create a new type of accident or malfunction, and the method and manner of plant operation remains unchanged. This change will not alter the safety functions of the rod control system. The safety design bases in the USAR have not been altered, and no new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this change. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

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

There are no changes being made to any safety limits or safety system settings that would adversely impact plant safety. This proposed change will have no affect on the availability, operability or performance of any safety-related system or component. The change will not prevent inspections or surveillances required by the technical specifications, and does not alter the rod drop time criterion specified in the technical specifications. Performance of the rod drop tests at other temperatures allows an alternative method to verify that the rod drop time currently specified in the technical specifications and used in the safety analyses continues to be valid. Therefore, the proposed change would not result in a reduction in a margin of safety.

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

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

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

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

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

Date of amendment request: November 21, 1995

Brief description of amendment request: The proposed amendments would revise surveillance requirements for the high pressure coolant injection and reactor core isolation cooling systems and would make an administrative change to Section 5.5.7 of the technical specifications to eliminate reference to a section which was previously eliminated.

Date of publication of individual notice in Federal Register: December 5, 1995 (60 FR 62271).

Expiration date of individual notice: January 3, 1996.

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

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

Date of amendment request: November 30, 1995.

Brief description of amendment request: The proposed amendments would revise the minimum allowable control rod scram accumulator pressure and charging water header pressure from a value of 955 psig to a value of 940 psig.

Date of publication of individual notice in Federal Register: December 8, 1995 (60 FR 63073).

Expiration date of individual notice: January 8, 1996.

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

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

Date of amendment request: December 19, 1995.

Brief description of amendment request: The proposed amendment would revise the ventilation filter test program (VFTP) bypass and penetration leakage test acceptance criteria from less than 0.05 percent to less than 1.0 percent. The change corrects an administrative error that occurred during the development of the Peach Bottom Improved Technical Specifications which were issued as Amendments 210 and 214 to the Peach Bottom licenses on August 30, 1995.

Date of publication of individual notice in Federal Register: December 27, 1995 (60 FR 66997).

Expiration date of individual notice: January 25, 1996.

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

Notice of Issuance of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: October 20, 1995.

Brief description of amendment: The one-time amendment revises the Calvert Cliffs Nuclear Power Plant, Unit No. 1 Technical Specifications by extending certain 18-month instrument surveillance intervals by a maximum of 39 days to March 31, 1996. This amendment will be superseded by Amendment No. 208 when it is implemented prior to restart from the Unit No. 1 spring 1996 refueling outage.

Date of issuance: December 28, 1995.

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

Amendment No.: 209.

Facility Operating License No. DPR–53: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58396).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 2, 1995.

Brief description of amendment: The amendment revises the Technical Specifications regarding allowable outage time (AOT) associated with the control room emergency ventilation system. It extends the AOT for one train from 7 days to 30 days on a one-time basis (for the loss of the emergency power supply only) to allow for modifications during the upcoming Unit No. 1 refueling outage in the spring of 1996.

Date of issuance: December 19, 1995.

Effective date: As of the date of issuance to be implemented during the Unit No. 1 spring 1996 refueling outage.

Amendment No.: 187.

Facility Operating License No. DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56363).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No

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

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 amendments: September 10, 1993, as supplemented on June 16, 1995.

Brief description of amendments: This application upgrades the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specification General Electric Plants BWR/4.'' This application upgrades only Section 3/4.8 (Plant Systems).

Date of issuance: December 19, 1995.

Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 144, 138, 166, and 162.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37086).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No.

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.

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 amendments: September 15, 1995.

Brief description of amendments: The amendments upgrade the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specification General Electric Plants BWR/4.'' The application dated September 15, 1995, contains some of the TSUP open items from previous Dresden and Quad Cities TS amendments issued by the NRC.

Date of issuance: December 19, 1995.

Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 145, 139, 167 and 163

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 5, 1995 (60 FR 52220).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No.

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.

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 amendments: September 17, 1993, as supplemented July 28, 1995.

Brief description of amendments: This application upgrades the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specification General Electric Plants BWR/4.'' This application upgrades only Section 3/4.5 (Emergency Core Cooling Systems).

Date of issuance: December 27, 1995.

Effective date: Immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 146, 140, 168, and 164.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42599).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 1995.

No significant hazards consideration comments received: No.

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.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois.

Date of application for amendments: November 14, 1995.

Brief description of amendments: These amendments change the implementation dates of all previous TSUP amendments from December 31, 1995, to no later than June 30, 1996.

Date of issuance: December 29, 1995.

Effective date: December 29, 1995.

Amendment Nos.: 147 and 141.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the license.

Date of initial notice in Federal Register: November 29, 1995 (60 FR 61272).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 29, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Commonwealth Edison Company, Docket No. 50–373, LaSalle County Station, Unit 1, LaSalle County, Illinois.

Date of application for amendment: October 2, 1995.

Brief description of amendment: The amendment revises the safety/relief valve (SRV) safety function lift setting allowable tolerance band from −3/+1% to ±3% and includes a requirement for the lift settings to be within ±1% of the technical specification limit following testing.

Date of issuance: January 3, 1996.

Effective date: Upon date of issuance; shall be implemented prior to the restart of Unit 1 from its seventh refueling outage.

Amendment No.: 108.

Facility Operating License No. NPF–11: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58398).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 5, 1995.

Brief description of amendments: In Section 5.2.5 of the Catawba Safety Evaluation Report (SER, NUREG–0954), the NRC staff identified that the air particulate monitors (EMF38, at both Units 1 and 2), are designed to seismic Category I requirements. A recent engineering review by the licensee determined that documentation did not exist to show these monitors are designed to seismic Category I requirements. In a submittal dated September 8, 1994, the licensee proposed a technical justification for not requiring the subject monitors to be seismic Category I, and by letter dated September 5, 1995, provided additional justification and requested amendments to the licenses for both Units 1 and 2. The NRC staff has reviewed the licensee's justification and concludes that the containment air particulate monitors at Catawba do not have to meet seismic Category I requirements. The bases for this conclusion are included in the NRC staff's Safety Evaluation.

Date of issuance: December 29, 1995.

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

Amendment Nos.: Unit 1—140; Unit 2—134.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in Federal Register: November 28, 1995 (60 FR 58690).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 29, 1995 and an Environmental Assessment dated December 22, 1995.

No significant hazards consideration comments received: No.

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

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina.

Date of application for amendments: September 1, 1995, as supplemented by letters dated October 17 and November 15, 1995.

Brief description of amendments: The requested changes would revise Technical Specification (TS) 6.9.1.9 to include references to updated or recently approved methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report (COLR). The subject references have previously been reviewed and approved by the NRC staff.

Date of issuance: December 19, 1995.

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

Amendment Nos.: Unit 1—160; Unit 2—142.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 25, 1995 (60 FR 54718).

The October 17 and November 15, 1995, letters provided clarifying information that did not change the scope of the September 1, 1995, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina.

Date of application for amendments: January 12, 1995, as supplemented by letter dated June 29, 1995.

Brief description of amendments: The amendments would revise and clarify portions of Technical Specification Section 6.0, ''Administrative Controls.''

Date of issuance: December 19, 1995.

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

Amendment Nos.: Unit 1—161; Unit 2—143.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 15, 1995 (60 FR 14018).

The June 29, 1995, letter provided clarifying information that did not change the scope of the January 12, 1995, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

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 of amendments: July 26, 1995, as supplemented by letter dated November 20, 1995.

Brief description of amendments: The amendments add a footnote to Technical Specification 3.7.8 to provide for a one-time extension of the allowable outage time from 72 hours to 7 days for the Oconee overhead emergency power path to be inoperable, so that proposed modifications to the degraded grid protection system and the external grid trouble protection system may be performed.

Date of Issuance: December 27, 1995.

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

Amendment Nos.: Unit 1—213; Unit 2—213; Unit 3—210.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42601).

The November 20, 1995, letter provided clarifying information that did not change the scope of the July 26, 1995, application and the proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 19, 1995.

Brief description of amendment: The amendment reduced the requirements associated with the exercise frequency of control element assemblies from once per 31 days to once per 92 days.

Date of issuance: December 22, 1995.

Effective date: December 22, 1995, to be implemented within 30 days.

Amendment No.: 173.

Facility Operating License No. NPF–6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52929).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 4, 1995.

Brief description of amendment: The amendment revises surveillance requirements associated with the main turbine steam valves.

Date of issuance: December 22, 1995.

Effective date: December 22, 1995, to be implemented within 30 days.

Amendment No.: 174.

Facility Operating License No. NPF–6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35069).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1995.

No significant hazards consideration comments received: No.

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

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida.

Date of application for amendments: September 11, 1995, as supplemented by letter dated November 22, 1995.

Brief description of amendments: These amendments revise the emergency diesel generator testing requirements to incorporate the recommendations of Generic Letters 93–05 and 94–01.

Date of issuance: December 28, 1995.

Effective date: December 28, 1995.

Amendment Nos. 181 and 175.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52930).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

Date of application for amendments: December 2, 1994.

Brief description of amendments: The amendments replace Appendix B, ''Environmental Technical Specifications,'' with an Environmental Protection Plan (Nonradiological) and revise the Operating Licenses to reflect these changes.

Date of issuance: December 19, 1995.

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

Amendment Nos.: Unit 1—199; Unit 2—140.

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

Date of initial notice in Federal Register: January 4, 1995 (60 FR 502).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 13, 1995, as supplemented August 28 and October 27, 1995.

Brief description of amendment: The amendment modifies the Technical Specifications to allow use of laser-welded sleeves to repair defective steam generator tubes.

Date of issuance: January 4, 1996.

Effective date: January 4, 1996, with full implementation within 45 days.

Amendment No.: 205.

Facility Operating License No. DPR–58. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29877).

The August 28 and October 27, 1995, supplements provided clarifying information and updated Technical Specification pages. These supplements did not change the proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 4, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 31, 1995, as supplemented December 5, 1995.

Brief description of amendment: The amendment modifies the definition of HOT SHUTDOWN and COLD SHUTDOWN to specify that the definitions are not applicable during the performance of an inservice hydrostatic and leak test (IHLT). Technical Specification Section 3.6.B and 4.6.B is modified by adding Section 3.6.B.1.b and 4.6.B.1.b to identify the requirements that must be satisfied to consider the reactor in COLD SHUTDOWN during the performance of an IHLT. In addition, the amendment changes temperature specific requirements on several pages to mode or condition specific requirements; makes several editorial changes; and changes the associated Bases.

Date of issuance: December 29, 1995.

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

Amendment No.: 90.

Facility Operating License No. DPR–21. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49940).

The December 5, 1995, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 1, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to extend the interval for performance of selected surveillances to accommodate a 24-month fuel cycle. Specifically, this amendment changes the definition for a refueling interval, changes the BASES for surveillances that are performed at least once each fuel cycle and changes the surveillance frequencies for:

(1) The flow path tests of the boron injection system,

(2) The operability tests of the digital rod position indicatiors,

(3) The drop time of the full-length shutdown and control rods,

(4) The channel calibration of the loose-part detection system,

(5) The channel calibration of the seismic monitoring instrumentation,

(6) The activation of the pumps and the flow path tests of the valves in the containment quench and recirculation spray systems and

(7) The tests of the intended actuation positions of the containment isolation valves.

Date of issuance: December 28, 1995.

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

Amendment No.: 122.

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58402).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 17, 1995.

Brief description of amendment: The amendment revises the Technical Specifications pertaining to the plant air filtration and ventilation systems to extend the surveillance frequencies that are now required to be performed at least once per 18 months to specify that the surveillances are to be performed at least once each refueling interval.

Date of issuance: December 28, 1995.

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

Amendment No.: 123.

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58402).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 14, 1995.

Brief description of amendment: The amendment revises the frequency of those surveillance requirements for the emergency core cooling systems that now require that the surveillances be performed ''at least once per 18 months'' to specify that the surveillances be performed ''at least once each refueling interval.''

Date of issuance: December 28, 1995.

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

Amendment No.: 124.

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58402).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 29, 1995.

Brief description of amendments: The amendments added a one-time footnote to the Technical Specifications related to the diesel generator fuel oil storage and transfer system to permit each of the existing storage tanks to be removed from service for up to 60 days so they can be replaced with double walled tanks and piping that comply with new California regulations.

Date of issuance: January 3, 1996.

Effective date: January 3, 1996, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 109; Unit 2—Amendment No. 108.

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58403).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 1996.

No significant hazards consideration comments received: No.

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

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California.

Date of application for amendment: October 8, 1993, as supplemented October 28, 1994.

Brief description of amendment: This amendment revised the Technical Specification by deleting Figure II–2, ''Restricted Area Per 10 CFR 20.3(a)(14)'' and by deleting the restricted area boundary line from Figure V–3, ''HBPP Groundwater Monitoring System Wells.''

Date of issuance: December 21, 1995.

Effective date: This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

Amendment No.: 30.

Facility License No. DPR–7: This amendment revised the TS.

Date of initial notice in Federal Register: January 5, 1994 (59 FR 624).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Humboldt County Library, 1313 3rd Street, Eureka, California 95501.

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

Date of application for amendments: March 31, 1995.

Brief description of amendments: The amendments incorporate a change in the Station Technical Specifications for both units that modifies the requirement in TS 4.4.4.3.a to have the pH of the reactor coolant measured every 72 hours. The amendments add the clarification that the pH measurement will be performed only when the coolant conductivity is greater than 1.0 micro-mho/cm at 25°C (°77).

Date of issuance: January 3, 1996.

Effective date: Both units, as of date of issuance and are to be implemented within 30 days.

Amendment Nos.: 156 and 127.

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

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20522).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 11, 1995.

Brief description of amendment: The amendment revises the Unit 2 Technical Specifications (TSs) to reestablish the original operability requirements for the Neutron Flux function, and to delete the footnote that was added to TS page 3/4 3–71 under Amendment No. 115, regarding the length of time that the revised operability values were valid.

Date of issuance: January 3, 1996.

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

Amendment No.: 128.

Facility Operating License No. NPF–22. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47623).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 12, 1995.

Brief description of amendment: The amendment modifies the Technical Specifications (TSs) to extend the surveillance test intervals for the emergency service water system to support 24-month operating cycles. Surveillance test interval extensions are denoted as being performed ''every 24 months'' or ''at least once per 24 months'' consistent with the guidance provided in Generic Letter (GL) 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate 24–Month Fuel Cycle,'' dated April 2, 1991. The NRC staff has determined that the proposed TS changes are in accordance with GL 91–04, and are therefore acceptable.

Date of issuance: December 21, 1995.

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

Amendment No.: 230.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register*:* September 13, 1995 (60 FR 47623)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1995.

No significant hazards consideration comments received: No.

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

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina.

Date of application for amendment: February 21, 1995, as supplemented on August 31, 1995, and December 4, 1995.

Brief description of amendment: The amendment revises the Technical Specifications (TS) support of the licensee's plan to implement the revised 10 CFR Part 20, ''Standards for Protection Against Radiation.'' Also, several editorial changes to improve the clarity of the TS were made.

Date of issuance: December 28, 1995.

Effective date: 90 days after issuance.

Amendment No.: 130.

Facility Operating License No. NPF–12. Amendment revises the operating license.

Date of initial notice in Federal Register*:* March 29, 1995 (60 FR 16200). Renoticed on September 27, 1995 (60 FR 49946) due to changes in the licensee's proposed no significant hazards consideration analysis that were included in the August 31, 1995 supplemental letter. The December 4, 1995 letter provided supplemental information that did not change the second proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

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

Date of application for amendment: June 21, 1994, as supplemented by letter dated October 23, 1995.

Brief description of amendment: The amendment revises Technical Specification (TS) 6.5.1, 6.5.2 and 6.5.3 to relocate the review and audit requirements of the On-site Review Committee (ORC) and the Nuclear Safety Review Board (NSRB) to the Operational Quality Assurance Manual (OQAM). In addition, the amendment deletes reference to the Manager, Nuclear Safety and Emergency Preparedness, in TS 6.2.3. The Index is revised to reflect the relocations.

Date of issuance: December 26, 1995.

Effective date: December 26, 1995, to be implemented within 30 days from the date of issuance.

Amendment No.: 107.

Facility Operating License No. NPF–30. The amendment revises the Technical Specifications.

Date of initial notice in Federal Register*:* August 31, 1994 (59 FR 45036) and November 27, 1995 (60 FR 58406). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 26, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 20, 1995.

Brief description of amendments: These amendments establish a new setpoint for the steam generator high-high level and provide more restrictive setting limits for certain reactor protection system/engineered safety features actuation system setpoints.

Date of issuance: December 28, 1995.

Effective date: December 28, 1995.

Amendment Nos. 206 and 206.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45190).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 19, 1995.

Brief description of amendment: The amendment makes administrative changes to the KNPP Technical Specifications (TS) to improve their clarity and consistency. The amendment includes changes to reflect revisions to 10 CFR Part 20, and changes to correct minor typographical and format inconsistencies as part of the licensee's ongoing effort to convert the TS to the WordPerfect format.

Date of issuance: December 21, 1995.

Effective date: December 21, 1995.

Amendment No.: 122.

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

Date of initial notice in Federal Register*:* October 11, 1995 (60 FR 52936).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1995.

No significant hazards consideration comments received: No.

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

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin.

Date of application for amendments: April 27, 1995, as supplemented by letter dated November 29, 1995.

Brief description of amendments: The amendments revise TS Table 15.3.5–1, ''Engineered Safety Features Initiation Instrument Setting Limits,'' and TS Table 15.3.5–3, ''Engineered Safety Features.'' Setting limits are modified and references are changed. The bases for TS Section 15.3.5, ''Instrumentation System,'' are also changed to be consistent with the TS changes.

Date of issuance: December 27, 1995.

Effective date: December 27, 1995.

Amendment Nos.: 167 and 171.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register*:* May 23, 1995 (60 FR 27346). The November 29, 1995, submittal provided supplemental information which did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Dated at Rockville, Maryland, this 11th day January 1996.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 96–676 Filed 1–19–96; 8:45 am]

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Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision to a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified by its task number, DG–8016 (which should be mentioned in all correspondence concerning this draft guide), is a proposed Revision 1 to Regulatory Guide 8.37, ''Constraints for Air Effluents for Licensees Other Than Power Reactors.'' This guide is being revised to provide guidance on demonstrating compliance with proposed constraints for air effluents. These constraints were delineated in amendments that were proposed for 10 CFR Part 20, ''Standards for Protection Against Radiation,'' on December 13, 1995 (60 FR 63984).

This draft guide is being issued to involve the public in the early stages of the development of a regulatory position in this area. It has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the draft guide. Comments should be accompanied by supporting data. Written comments may be submitted to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Comments will be most helpful if received by March 12, 1996.

Comments may be submitted electronically, in either ASCII text or Wordperfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FedWorld. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet. Background documents on the rulemaking are also available for downloading and viewing on the bulletin board.

If using a personal computer and modem, the NRC subsystem on FedWorld can be accessed directly by dialing the toll free number: 1–800–303–9672. Communication software parameters should be set as follows: parity to none, data bits to 8, and stop bits to 1 (N,8,1). Using ANSI or VT–100 terminal emulation, the NRC NUREGs and RegGuides for Comment subsystem can then be accessed by selecting the ''Rules Menu'' option from the ''NRC Main Menu.'' For further information about options available for NRC at FedWorld consult the ''Help/ Information Center'' from the ''NRC Main Menu.'' Users will find the ''FedWorld Online User's Guides'' particularly helpful. Many NRC subsystems and databases also have a ''Help/Information Center'' option that is tailored to the particular subsystem.

The NRC subsystem on FedWorld can also be accessed by a direct dial phone number for the main FedWorld BBS: 703–321–8020; Telnet via Internet: fedworld.gov (192.239.93.3); File Transfer Protocol (FTP) via Internet: ftp.fedworld.gov (192.239.92.205); and World Wide Web using: http://www.fedworld.gov (this is the Uniform Resource Locator (URL)).

If using a method other than the toll free number to contact FedWorld, the NRC subsystem will be accessed from the main FedWorld menu by selecting the ''F—Regulatory, Government Administration and State Systems,'' then selecting ''A—Regulatory Information Mall.'' At that point, a menu will be displayed that has an option ''A—U.S. Nuclear Regulatory Commission'' that will take you to the NRC Online main menu. You can also go directly to the NRC Online area by typing ''/go nrc'' at a FedWorld command line. If you access NRC from