will use their best efforts to keep each other informed of proposed changes in their respective rules and regulations and licensing, inspection and enforcement policies and criteria, and to obtain the comments and assistance of the other party thereon.

Article VII

The Commission and the Commonwealth agree that it is desirable to provide reciprocal recognition of licenses for the materials listed in Article I licensed by the other party or by any other Agreement State. Accordingly, the Commission and the State agree to use their best efforts to develop appropriate rules, regulations, and procedures by which such reciprocity will be accorded.

Article VIII

The Commission, upon its own initiative after reasonable notice and opportunity for hearing to the Commonwealth, or upon request of the Governor of the Commonwealth, may terminate or suspend all or part of this Agreement and reassert the licensing and regulatory authority vested in it under the Act if the Commission finds that (1) such termination or suspension is required to protect public health and safety, or (2) the Commonwealth has not complied with one or more of the requirements of Section 274 of the Act. The Commission may also, pursuant to Section 274j of the Act, temporarily suspend all or part of this Agreement if, in the judgement of the Commission, an emergency situation exists requiring immediate action to protect public health and safety and the Commonwealth has failed to take necessary steps. The Commission shall periodically review this Agreement and actions taken by the Commonwealth under this Agreement to ensure compliance with Section 274 of the Act.

Article IX

This Agreement shall become effective on [April 24, 1996,] >(date to be determined)< and shall remain in effect unless and until such time as it is terminated pursuant to Article VIII.

Done at [Boston, Massachusetts] >(location to be determined)<, in triplicate, this [24]th Day of [April, 1996] >(date to be determined)<.

For the United States Nuclear Regulatory Commission.

Shirley Ann Jackson,

Chairman.

For the Commonwealth of Massachusetts.

William F. Weld,

Governor.

[FR Doc. 96–33252 Filed 12–31–96; 8:45 am]

BILLING CODE 7590–01–P

Sunshine Act Meeting

DATE: Weeks of December 30, 1996 and January 6, 13, and 20, 1997.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of December 30

There are no meetings scheduled for the Week of December 30.

Week of January 6–Tentative

Tuesday, January 7

9:30 a.m. Briefing on Investigative Matters (Closed—Ex. 5 & 7)
2:00 p.m. Discussion of Procedures for NRC Strategic Assessment (Closed—Ex. 2)

Thursday, January 9

10:00 a.m. Briefing by Maine Yankee, NRR, and Region I (PUBLIC MEETING) (Contact: Daniel Dorman, 301–415–1429)
12:00 am. Affirmation Session (PUBLIC MEETING) (if needed)

Week of January 13–Tentative

Monday, January 13

10:00 a.m. Briefing on NRC Strategic Assessment (PUBLIC MEETING) (Contact: John Craig, 301–415–3812)
11:30 a.m. Affirmation Session (PUBLIC MEETING) (if needed)

Week of January 20–Tentative

Tuesday, January 21

3:30 p.m. Briefing on Investigative Matters (Closed—Ex, 5 & 7)

Wednesday, January 22

10:00 a.m. Briefing on Codes and Standards (PUBLIC MEETING) (Contact: Gil Millman, 301–415–5843)
11:30 a.m. Affirmation Session (PUBLIC MEETING) (if needed)

*THE SCHEDULE FOR COMMISSION MEETINGS IS SUBJECT TO CHANGE ON SHORT NOTICE. TO VERIFY THE STATUS OF MEETINGS CALL (RECORDING)—(301) 415–1292. CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661).

In addition, distribution of this meeting notice over the internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

* * * * *

Dated: December 27, 1996.

[FR Doc. 97–00063 Filed 12–30–96; 12:45 pm]

BILLING CODE 7590–01–M

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 9, 1996, through December 19, 1996. The last biweekly notice was published on December 18, 1996.

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

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

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

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

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

By February 3, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendments request: November 6, 1996

Description of amendments request: The proposed amendment would modify the technical specifications (TS) to require manual blocking of one train of fast bus transfer (FBT) within the first hour of degraded switchyard voltage should the switchyard voltage fall below the level necessary for the electrical distribution system (EDS) degraded voltage protection to maintain compliance with General Design Criteria (GDC) 17. The proposed amendment would further require the starting, paralleling with the grid, loading, and then separating from the grid the other train's emergency diesel generator (EDG) within the first hour, rather than the current TS which allows two hours after onset of a degraded switchyard voltage condition to start the EDG. Alternatively, fast bus transfer can be blocked in both trains within the first hour. The proposed amendment includes changes to the applicable notes to reflect that these changes are no longer temporary, but will remain as part of the long-term solution to this issue.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration. The NRC staff's analysis 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 reduces the amount of time the second train of electrical equipment is allowed to remain in nonconformance with GDC 17 in the TS action statement. This change only affects equipment used to mitigate an event, and does not affect equipment assumed to initiate any event. Thus the probability of an accident previously evaluated is not affected.

The proposed change brings the second EDS train into compliance with GDC 17 at least one hour sooner than the current TS. Once in conformance with GDC 17, the consequences of accidents previously evaluated conform to the current analysis. Thus the proposed change does not 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 only affects equipment designed to mitigate the effects of an accident. The proposed change ensures that safety equipment is configured as assumed in the current accident analysis. The proposed change does not affect the conditions of structures, systems, or components assumed in the safety analysis beyond the existing design basis as maintained by the current TS. The proposed change does not, therefore, 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 margin of safety affected by the proposed change is based on calculated offsite dose consequences for postulated transients and accidents for which the EDS provides power for equipment required to mitigate. The proposed change reduces the time that one train of the EDS is allowed to remain in nonconformance with GDC 17, thus increasing the availability of the EDS prior to the onset of a postulated accident compared to the current TS. Thus the proposed change does not increase thecalculated offsite dose, and therefore the proposed change does not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: November 26, 1996

Description of amendments request: The proposed amendment will adopt Option B of 10 CFR Part 50, Appendix J, to require Type B and Type C containment leakage rate testing to be performed on a performance-based testing schedule. Containment leakage rate testing is currently performed in accordance with 10 CFR Part 50, Appendix J, Option A, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' Appendix J specifies containment leakage testing requirements, including the types of tests required, frequency of testing, and reporting requirements. Containment leakage test requirements include performance of Integrated Leakage Rate Tests, also known as Type A tests, which measure overall leakage rate of the containment; and Local Leakage Rate Tests, also known as Types B and C tests, which measure the leakage through containment penetrations and valves. The Nuclear Regulatory Commission (NRC) has amended the regulations to provide an alternate performance-based option, Option B, to the existing Appendix J. Baltimore Gas and Electric Company (BGE) received approval to adopt Option B for Type A testing only. At this time, BGE plans to adopt Option B for Types B and C testing, as well.

BGE is revising the Containment Leakage Rate Testing Program for Type A testing to implement Types B and C testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. The revised program will be developed in accordance with the guidelines contained in Regulatory Guide 1.163 ''Performance-Based Containment Leak-Rate Test Program,'' dated September 1995, including errata.

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.

Containment leakage rate testing is performed in accordance with 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' The Appendix J containment leakage test requirements include performance of Type A tests, which measure the overall leakage rate of the containment, and Types B and C tests, which measure the leakage through containment penetrations and valves. The Nuclear Regulatory Commission has amended the regulations to provide a performance-based alternative, Option B, to the existing Appendix J. Baltimore Gas and Electric Company adopted Option B for Type A testing during the Unit 1 refueling outage earlier this year. At this time, BGE plans to adopt Option B for Types B and C testing.

Implementation of Option B involves no physical or operational changes to the plant structures, systems, or components. Furthermore, leakage rate does not contribute to the initiation of any postulated accidents; therefore, this proposed change does not involve an increase in the probability of any previously evaluated accidents.

Types B and C testing is necessary to demonstrate that leakage through the containment penetrations is within the limits assumed in the accident analyses. The only potential effect of the proposed change to the Types B and C test frequency is the possibility that containment penetration leakage would go undetected between tests. To provide assurance that containment penetration leakage remains within the limits of the Technical Specifications, BGE plans to implement the performance-based leakage testing program in accordance with NRC Regulatory Guide 1.163, dated September 1995 (including errata), with no exceptions.

By adopting Option B, BGE will no longer require an exemption from 10 CFR Part 50, Appendix J, which was granted to accommodate 24-month operating cycles. The exemption increased the surveillance interval to a maximum of 30 months, while proportionately decreasing the combined Types B and C leakage rate acceptance criteria. Option B to Appendix J provides the regulation necessary to accommodate an extended fuel cycle, while maintaining the original combined Types B and C leakage rate testing limit. Therefore, BGE has requested revocation of the exemption to 10 CFR Part 50, Appendix J, as adoption of Option B for Types B and C testing will enable a return to full compliance with Appendix J. As the facility will be in full compliance with the regulations, this change does not increase the consequences of any previously evaluated accidents.

Implementation of Option B does not change the total allowable containment leakage rate acceptance criteria, nor does it change the total leakage assumed in the accident analyses. Option B allows the implementation of a performance-based testing program to ensure that resources are concentrated on the components most likely to exceed administrative limits. Similarly, the changes to relocate the procedural details, including test frequency, performance and data conversion methodology, for containment leakage rate testing from the Technical Specifications to the Containment Leakage Rate Testing Program will have no effect on the total containment leakage allowed by the Technical Specifications, or assumed in the accident analyses. Relocating the allowable leakage rate conversions (Standard Cubic Centimeters per Minute) to the Technical Specification Bases does not change the allowable leakage rates (as a percentage of the containment air volume) specified in the Technical Specifications. Furthermore, relocation of the programmatic controls for Types B and C testing, including the allowable leakage rates, to the Administrative Controls section of the Technical Specifications ensures an adequate level of regulatory control of these criteria is retained.

Additionally, the Calvert Cliffs Individual Plant Examination considered the effects associated with severe accidents which could lead to containment failure. It was concluded that adopting a performance-based testing interval will not significantly affect the containment failure probabilities calculated for the Individual Plant Examination. Altogether, adoption of a performance-based testing frequency, as specified in 10 CFR Part 50, Appendix J, Option B, will not significantly decrease the confidence in the leak-tightness of the containment, including containment penetrations. Therefore, this change will not result in a significant increase in the probability of undetected containment penetration leakage in excess of that allowed by the Containment Leakage Rate Testing Program, or assumed in the accident analysis, or in the consequences of an accident previously evaluated.

Therefore, the proposed 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 adopts a performance-based approach to containment penetration leakage rate testing. This change does not add any new equipment, modify any interfaces with any existing equipment, or change the equipment's function, or the method of operating the equipment. The proposed change does not affect normal plant operations or configuration, nor does it affect leakage rate test methods. As the proposed change would not change the design, configuration or operation of the plant, it could not cause containment penetration leakage rate testing to 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 purpose of the existing schedule for Types B and C tests is to provide assurance, on a regular basis, that the release of radioactive material will be restricted to those leak paths and leakage rates assumed in the accident analyses. The margin of safety associated with containment penetration leakage rates is not reduced if containment leakage does not exceed the maximum allowable leakage rate defined in the Technical Specifications. Implementation of Option B does not change the total allowable containment leakage rate acceptance criteria, nor does it change the total leakage assumed in the accident analyses. Option B only allows the implementation of a performance-based testing program to ensure that resources are concentrated on the components most likely to exceed administrative limits. Similarly, the changes to relocate the procedural details for containment leakage rate testing from the Technical Specifications to either the Containment Leakage Rate Testing Program or the Technical Specification Bases will have no effect on the total containment leakage allowed by the Technical Specifications, or assumed in the accident analyses. Furthermore, relocation of the programmatic controls for Types B and C testing, including the allowable leakage rates, to the Administrative Controls section of the Technical Specifications ensures that the same regulatory control of these criteria is retained.

Elimination of the exemption to Appendix J which reduced the amount of combined Types B and C testing allowable leakage redistributes that portion of the total containment leakage which may be attributed to local leakage rate testing, but does not affect the maximum allowable containment leakage rate, L_a. The proposed change does not affect a safety limit, a Limiting Condition for Operation, or the way in which the plant is operated.

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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: S. Singh Bajwa, Acting Director

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

Date of amendment request: December 2, 1996 (NRC-96-0134)

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.1.4.3, Rod Block Monitor, and Tables 3.3.6-1 and 4.3.6-1 in TS 3.3.6, Control Rod Block Instrumentation, to expand the range of conditions under which the rod block monitor must be operable. These changes are required to ensure that all fuel limits are met for the core that has been loaded for Cycle 6.

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

1. The proposed changes provide requirements that are more restrictive than the existing requirements for operation of the facility. These changes provide assurance that the Rod Block Monitor system is operable when necessary to prevent or mitigate transients that could potentially threaten the integrity of the fuel cladding. There will be no adverse impact on the probability of any accident previously evaluated since the change provides additional assurance that fuel thermal and mechanical design bases will be satisfied and has no effect on any accident initiating mechanism. The additional restrictive conditions on plant operation also ensure that the consequences of anticipated operational occurrences are no more severe than the most limiting conditions using the current Technical Specifications. Therefore these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in Rod Block Monitor operability requirements are consistent with the current safety analysis assumptions. These requirements provide assurance that the Rod Block Monitor will be operable if necessary to terminate a rod withdrawal error so that fuel thermal and mechanical design limits are satisfied. The change does not cause a physical change to the plant or introduce a new mode of operation. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. These changes maintain current assumptions within the safety analyses and design basis. The changes provide assurance that the Rod Block Monitor will be operable if necessary to terminate a rod withdrawal error so that fuel thermal and mechanical design limits are satisfied. Therefore, these changes do not involve a reduction in a margin of safety.

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

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226

NRC Project Director: John N. Hannon

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: November 6, 1996

Description of amendment request: The proposed amendment would revise the technical specifications to permit an increase in the allowable leak rate for the Main Steam Isolation Valves (MSIVs) and delete the Penetration Valve Leakage Control System (PVLCS) and Main Steam-Positive Leakage Control System (MS-PLCS) requirements.

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

(1) The operation of River Bend Station, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 involves eliminating the PVLCS and MS-PLCS leakage control requirements from the Technical Specifications. As described in Sections 9.3 and 6.7 respectively, of the USAR [Updated Safety Analysis Report], the PVLCS and MS-PLCS are manually initialed about 20 minutes following a design basis LOCA [Loss of Coolant Accident].

Since the PVLCS and MS-PLCS are operated only after an accident has occurred, this proposed amendment has no effect on the probability of an accident.

Since MSIV leakage and operation of the PVLCS and MS-PLCS are included in the radiological analysis for the design basis LOCA as described in Section 15.6.5 of the USAR, the proposed amendments will not affect the precursors of other analyzed accidents. The PVLCS and MS-PLCS are not initiators of any previously analyzed accident. The proposed amendments result in acceptable radiological consequences of the design basis LOCA previously evaluated in Section 15.6.5 of the USAR.

The proposed amendment to Technical Specification 3.6.1.3 does not involve a change to structures, components or systems that would affect the probability of an accident previously evaluated. A plant-specific radiological analysis has been performed to assess the affects of the proposed increase to the allowable MSIV leak rate and deletion of the PVLCS and MS-PLCS in terms of Control Room and off-site doses following a postulated design basis LOCA. This change required a revision to the existing LOCA dose analysis due to the potential leakage from the MSIVs and those valves served by the PVLCS. Additional changes were also included in the revised dose analysis to account for changes in regulatory guidance and dose methodology. Leakage from the drywell to the atmosphere through the PVLCS (secondary containment bypass valves) are both assumed to begin at time zero. The model conservatively assumes that one inboard MSIV fails open at time zero and the MSIVs associated with the remaining three main steam lines are assumed to begin leakage at 2 hours with a total leak rate of 200 scfh for all four main steam lines. The design basis leak rate of the primary containment (excluding main steam lines and lines sealed by the PVLCS) is 0.26% of the containment volume by weight per 24 hours for the duration of the accident and is assumed to be released entirely to the environment initially or the secondary containment later into the accident. The leakage of 170,000 cc/hr (4298 sccm) at P_a through the containment isolation valves served by the PVLCS is considered as bypass leakage circumventing the secondary containment. The on-site and off-site doses were determined using the TRANSACT computer code which included the ICRP 30 dose conversion factors. The total off-site and on-site LOCA doses for both the airborne and liquid release pathways resulting from the proposed change are bounded by the applicable regulatory limits.

The analysis demonstrates that dose contributions from the proposed combined MSIV leakage rate limit of 200 scfh and from the proposed deletion of the PVLCS and MS-PLCS result in values bounded by the applicable regulatory limits as compared to the LOCA doses previously evaluated for the off-site and Control Room doses as contained in 10CFR100 and 10CFR50, Appendix A (General Design Criteria 19), respectively. The LOCA doses previously evaluated are discussed in Section 15.6.5 of the USAR.

The whole body (DDE [Deep Dose Equivalent]) doses at the Low Population Zone (LPZ) is 2.82 Rem and the Control Room is 0.43 Rem. These values are acceptable since the revised doses are bounded by the Regulatory Guidelines (2.82 versus 25 Rem at the LPZ and 0.43 versus 5 Rem at the Control Room). The associated whole boy (DDE) dose at the exclusion area boundary (EAB) is 4.69 Rem which also remains bounded by the Regulatory Guideline of 25 Rem.

The thyroid CEDE [Committed Effective Dose Equivalent] dose at the LPZ is 62.58 Rem. This is acceptable since the revised dose of 62.58 Rem is significantly less than the Regulatory Guideline (300 Rem). The EAB thyroid CEDE dose is 37.53 Rem, whereas the Control Room thyroid CEDE dose is 11.18 Rem. These values are also acceptable since the revised doses are well within the Regulatory Guidelines (37.53 versus 300 Rem at the EAB and 11.18 versus 30 Rem at the Control Room). The Control Room beta (SDE [Shallow Dose Equivalent]) dose is 9.15 Rem which also remains bounded by the Regulatory Guideline of 30 Rem.

In summary, the proposed changes do not result in an increase to the radiological consequences of a LOCA previously evaluated in the USAR. The revised LOCA doses are bounded by the Regulatory Guidelines. The effectiveness of the proposed request even for leakage rates greater than the proposed MSIV allowable leak rate ensures that off-site and Control Room dose limits are not exceeded.

There is no physical change to the ADS/SRVs [Automatic Depressurization System/Safety Relief Valve]. The PVLCS accumulator tanks remain the backup air supply to the ADS/SRV accumulators. A qualified long-term backup air supply remains but is supplied from a difference source. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deletes the requirements for the LCS [Leakage Control System] isolation valves which are non-PCIVs. These valves are eliminated and will not be performing a safety function. The LCS lines that are connected to the PCIVs and process piping will be welded and/or capped closed to assure primary containment integrity is maintained. The welding and post-weld examination procedures will be in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI requirements. These welds and/or caps will be periodically tested as part of the primary Containment Integrated Leak Rate Test (CILRT) program in accordance with the requirements of 10CFR50, Appendix J. The proposed change does not involve an increase in the probability of equipment malfunction previously evaluated in the USAR. In fact, the proposed change reduces the probability of equipment malfunction since, upon implementation, RBS will be operated with fewer process line isolation valves and associated support equipment subjected to postulated failure. The affected LCS MOVs [Motor Operated Valves] will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. This proposed change has no effect on the consequences of an accident previously evaluated since the LCS lines will be welded and/or capped closed, thus assuring that primary containment integrity, isolation and leak test capability are not compromised.

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

(2) The operation of River Bend Station, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment to Technical Specification 3.6.1.3 does not create the possibility for a new or different kind of accident from any accident previously evaluated. The BWROG (Boiling Water Reactors Owners Group) evaluated MSIV leakage performance and concluded that MSIV leakage rates up to 200 scfh will not inhibit the capability and isolation performance of the valve to isolate the primary containment. There is no new modification which could impact the MSIV operability. The LOCA has been reanalyzed at the proposed maximum combined leakage rate of 200 scfh. Therefore, the proposed change does not create any new or different kind of accident from any accident previously evaluated in the USAR.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the PVLCS and MS-PLCS does not affect any of the remaining systems at RBS [River Bend Station) and the LOCA has been reanalyzed with LOCA doses resulting from the proposed change remaining bounded by the applicable regulatory limits.

The PVLCS and MS-PLCS are of low safety significance as discussed in NUREG-1273, Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3, ''Containment Integrity Check,'' and NUREG/CR-3539, ''Impact of Containment Building Leakage on LWR Accident Risk.''

The proposed change to eliminate the LCS does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the LCS does not adversely affect any of the remaining RBS systems or change system inter-relationships. The associated proposed changes to delete the LCS isolation valves does not create the possibility of a new or different kind of accident. The affected LCS MOVs will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. The PVLCS and MS-PLCS connections to the process piping will be welded and/or capped closed to assure that primary containment integrity, isolation and leak testing capability are not compromised, therefore eliminating the possibility for any new or different kind of accident.

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

(3) The operation of River Bend Station, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed amendment to Technical Specification 3.6.1.3 does not involve a significant reduction in a margin of safety. The allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis demonstrate calculated doses, assuming the two single active failures of one MSIV to close and one diesel generator to respond are bounded by the requirements of 10CFR100 for the off-site doses and 10CFR50, Appendix A (General Design Criteria 19) for the Control Room doses. The calculated whole body doses are significantly reduced at the LPZ, the Control Room, and the EAB. The calculated thyroid dose is significantly reduced at the LPZ, the Control Room, and the EAB.

The proposed amendment to delete Technical Specification 3.6.1.8 and 3.6.1.9 for the PVLCS and MS-PLCS, does not reduce the margin of safety. In fact, the overall margin of safety is increased. The method is effective to reduce dose consequences of MSIV and the PVLCS leakage over an expanded operating range and will, thereby, resolve the safety concern that the PVLCS and MS-PLCS will not function at leakage rates higher than their design capacity. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the environment. Therefore, the proposed method is highly reliable and effective for MSIV leakage and deletion of the PVLCS and MS-PLCS.

The calculation shows that MSIV leakage rates up to 100 scfh per steam line would not exceed the regulatory limits. Therefore, the proposed method provides a substantial safety margin for mitigating the radiological consequences of MSIV leakage beyond the proposed Technical Specification leak rate limit of 200 scfh for all four main steam lines (combined maximum pathway).

Minor increases in containment leakage such as the leakage through the MSIVs, as identified in NUREG-1273, NUREG/CR-3539, and NUREG-1493 have been found to have no significant impact on the risk to the public. Therefore, the proposed change does not result in a significant reduction in a margin of safety.

The proposed change to delete the LCS isolation valves does not reduce the margin of safety. Welded and/or capped closure of the LCS lines assure that primary containment integrity and leak testing capability are not compromised. The affected LCS MOVs will be eliminated or retained as normal system isolation or maintenance valves having no safety or leakage control function thus requiring no bypassing of their thermal overloads. The PVLCS and MS-PLCS connections to the process piping will be welded and/or capped closed to assure that primary containment integrity, isolation and leak testing capability are not compromised, therefore eliminating the possibility for a significant reduction in the margin of safety.

Therefore, as discussed above, 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: November 15, 1996

Description of amendment request: The proposed amendment would revise the technical specifications to allow the performance of the 24-hour emergency diesel generator (EDG) maintenance run while the unit is in either Mode 1 or Mode 2. This test for the River Bend Station (RBS) is currently prohibited in Mode 1 and Mode 2 and allowed in Modes 3, 4, and 5.

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 significantly increase the probability or consequences of an accident previously evaluated.

The RBS SAR [Safety Analysis Report] assumes that the AC [Alternating Current] electrical power sources are designed to provide sufficient capacity, capability, redundancy and reliability to ensure that the fuel, reactor coolant system and containment design limits are not exceeded during an assumed design basis event. Specifically, the SAR assumes that the onsite EDGs provide emergency power in the event offsite power is lost to either one or all three EDF [Engineered Safety Features]

buses. In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the EDGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a design basis accident such as a LOCA [Loss of Coolant Accident].

The proposed change to permit the 24-hour testing of the EDGs during power operation does not significantly increase the probability or consequences of any previously evaluated accident. The capability of the EDGs to supply power in a timely manner will not be compromised by permitting performance of EDG testing during periods of power operation. Design features of the EDGs and electrical systems ensure that if a LOCA or LOP [Loss of Offsite Power] signal, either individually or concurrently, should occur during testing, the EDG would be returned to its ready-to-load condition (i.e., EDG running at rated speed and voltage separated from the offsite sources) or separately connected to the ESF bus providing ESF loads. An EDG being tested is considered to be operable and fully capable of meeting its intended design function. Additionally, the testing of an EDG is not a precursor to any preciously evaluated accidents.

If, during the test period, the EDG were to receive a normal operation protective trip resulting in the actuation of a generator lockout signal, the lockout could be reset by the operators monitoring the test. The resulting delay does not present an immediate challenge to the fuel cladding integrity, reactor water level control or to containment parameters, as demonstrated by the bounding four-hour station blackout coping analysis contained in RBS's station blackout conformance report.

Therefore, the proposed change allowing testing of EDGs during power operation will not significantly increase 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.

As previously discussed, the proposed change to permit the performance of EDG testing during power operation will not affect the operation of any system or alter any system's response to previously evaluated design basis events. The EDGs will automatically transfer from the test configuration to the ready-to-load configuration following receipt of a valid signal (i.e., LOCA or LOP). In the ready-to-load configuration the EDG will be running at rated speed and voltage, separated from the offsite source and capable of automatically supplying power to the ESF buses in the event that preferred power is actually lost.

The proposed change is also the same configuration currently used for the monthly one-hour test. Therefore, testing during power operation will not create the possibility of a new or different kind of event from any previously evaluated.

[Surveillance Requirement] SR 3.8.1.16 demonstrated that the EDG will automatically override the test mode following generation of a LOCA signal. In addition, the ability of the EDGs to survive a full load reject is verified by the performance of SR 3.8.1.9. These existing surveillance requirements, along with system design features, ensure that the performance of EDG testing during power operation will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded. Specifically, the EDGs must be capable of automatically providing power to ESF loads in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a design basis accident in the event of a loss of preferred power.

Testing of EDGs during power operation will not affect the availability or operation of any offsite source of power. In addition, the EDG being tested remains capable of meeting it intended design functions. Therefore, the proposed change to the Technical Specification Surveillance Requirement 3.8.1.13 will 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 location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

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NRC Project Director: William D. Beckner

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: November 15, 1996

Description of amendment request: The proposed amendment would increase the two recirculation loop Minimum Critical Power Ratio (MCPR) limit from 1.07 to 1.10 and the single recirculation loop MCPR limit from 1.08 to 1.12. This change request is the result of a non-conservative calculation identified by the fuel vendor.

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

The revised Safety Limit MCPR and the cycle-specific thermal limits that are based on the revised SLMCPR have been calculated using the methods identified in the ''Supplemental Reload Licensing Report For River Bend Station Reload 6 Cycle 7'' (Reference 1). These methods are within the existing design and licensing basis and cannot increase the probability or severity of an accident. The basis of the MCPR Safety Limit calculation is to ensure that greater that [than] 99.9% of all fuel rods in the core avoid transition boiling and fuel damage in the event of a postulated accident.

The SLMCPR is used to establish the Operating Limit Minimum Critical Power Ratio (OLMCPR). Neither the SLMCPR nor the OLMCPR can initiate an event, therefore[,] a change to the SLMCPR does not increase the probability of a accident previously evaluated. Maintaining the Minimum Critical Power Ratio (MCPR) at or above the OLMCPR during normal operations precludes fuel failure due to overheating of the fuel clad during an anticipated operational occurrence (AOO), thus limiting the consequences of an AOO. The proposed change will increase the SLMCPR, which will require the OLMCPR to be increased, which in turn will ensure that the requirements of 10 CFR [Part] 100 are met for an AOO. Therefore, there is no increase in the consequences of an accident previously analyzed.

The request does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

The MCPR Safety Limit is a Technical Specification numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. It cannot create the possibility of any new type of accident.

Neither the SLMCPR or the OLMCPR can initiate an event, therefore, a change to the SLMCPR does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

The request does not involve a significant reduction in the margin of safety.

The MCPR Safety Limit is a Technical Specification numerical value designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. This new Safety Limit MCPR is calculated using the methods identified in the reference. These methods are within the existing design and licensing basis and based on RBS specific inputs.

The margin of Safety resides between the SLMCPR and the point at which fuel fails. The proposed change to SLMCPR (and the OLMCPR) will in fact restore the margin of safety associated with GE's SLMCPR methodology.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

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NRC Project Director: William D. Beckner

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

Date of amendment request: December 2, 1996

Description of amendment request: The proposed Technical Specification (TS) Change Request will permit the use of 10CFR50 Appendix J, Option B, Performance-Based Containment Leakage Testing for Type A, B and C leak rate testing. TSs 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 4.6.1.6 and 4.6.1.7 are revised and Section 6.15 is added establishing the Containment Leakage Rate Testing Program. The Bases are revised to reflect this change. Minor editorial changes are included in this request. Waterford Steam Electric Station is planning to have a Containment Leakage Rate Testing Program in place prior to the next scheduled refueling outage. This program will be in accordance with the guidelines contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' dated September 1995.

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

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

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

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

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

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

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

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 amendment request: September 19, 1996

Description of amendment request: The proposed changes to Plant Hatch Units 1 and 2 Technical Specifications would revise the Surveillance Requirements (SRs) addressing the reactor vessel pressure and temperature (P/T) limits. The affected SRs are 3.4.9.1, 3.4.9.2, 3.4.9.3, 3.4.9.4, 3.4.9.5, 3.4.9.6, and 3.4.9.7, and the corresponding Units 1 and 2 Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3, which show P/T limit curves for inservice leak and hydrostatic testing, non-nuclear heatup and cooldown, and criticality, respectively.

The P/T curves would be changed to allow separate monitoring of the three major regions of the reactor pressure vessel (RPV) (i.e., the upper vessel and flange region, the beltline region, and the bottom head region), and to extend the validity of the Unit 1 curves to 32

Effective Full Power Years (EFPY). Separate monitoring would alleviate the difficulties with meeting certain temperature requirements due to the artificial limits imposed by the current P/T curves.

In support of the proposed changes, General Electric (GE) prepared and issued GENE-523-A137-1295, ''E. I. Hatch Nuclear Power Station, P-T Curve Modification for Unit 1 and Unit 2,'' which is provided in the submittal.

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

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

Pressure and temperature (P/T) limits for the reactor pressure vessel (RPV) are established to ensure brittle fracture of the vessel does not occur.

A. The proposed changes merely clarify the Applicability of the P/T limits for each of the low pressure conditions by replacing the word ''performed'' with ''met'', adding Notes to Surveillance Requirements, incorporating the requirements of Notes into the Surveillance Requirements, and modifying the Frequency statements. Conditions 2, 3, and 4, discussed in Enclosure 1 ''justification of changes'', [of the licensee's application] have their own Surveillance Requirements. Temperature requirements for Condition 1 are specified in the Bases. This proposed change only clarifies which Surveillance Requirement applies to each operating configuration. No reduction in Surveillance Frequencies is proposed.

B. The proposed revisions to the operating limits curves for inservice leak and hydrostatic testing, and the heatup and cooldown allow independent monitoring of the three RPV regions; i.e., the bottom head, the upper vessel and flange, and the core beltline. The three Unit 1 curves, including the criticality curve, were extended to 32 Effective Full Power Years (EFPY), and a correction to the Unit 1 criticality curve was made. Operating limits for each of the curves were evaluated in accordance with the methodology given in the applicable ASME Codes; Regulatory Guide 1.99, Rev. 2, and Appendix G of 10 CFR [Part] 50.

The actual limits in the inservice leak and hydrostatic testing curves, and the heatup and cooldown curves were not relaxed. Therefore, segregating the curves into the three affected vessel regions does not represent a reduction in the actual P/T requirements. The current P/T curves represent a composite of the three regions, with each point representing the limiting region. Regions of the vessel that are not limiting at a specific point are, therefore, artificially restrained. Upon implementation of the proposed changes, each vessel region will have its own curve, with its own true limit.

Since the proposed changes do not affect the recirculation piping, the probability and the consequences of a loss of coolant accident are not increased. Likewise, no other previously evaluated accidents or transients, as defined in Chapters 14 and 15 of the Units 1 and 2 Final Safety Analysis Reports, are affected by the proposed changes.

In summary, the proposed changes do not represent a relaxation of any actual operating limit and do not reduce the Frequency of any Surveillance. Three of the four operating configurations of the RPV are covered by Surveillance Requirements. Temperature limitations for the head removed from the vessel are given in the Bases. The operating limits were developed using the approved methodology contained in 10 CFR [Part] 50, Appendix G. Therefore, the probability and consequences of a brittle fracture of the RPV are not increased.

2. Do the proposed changes create the possibility of a new or different type of accident from any previously evaluated.

Implementing the low pressure changes, or the new operating limit curves, does not alter the design or operation of any system designed for the prevention or mitigation of accidents. The proposed changes do not introduce any new type of normal or abnormal operating mode or failure mode. All P/T limits for the Unit 1 and the Unit 2 reactor vessels continue to be monitored per the requirements of 10 CFR [Part] 50, Appendices G and H. Therefore, the proposed changes do not create the possibility of a new type of accident.

3. Do the proposed changes involve a significant reduction in the margin of safety?

The purpose of the P/T limits is to ensure a brittle fracture of the RPV does not occur. The proposed Technical Specifications changes for the low pressure conditions are made for clarification purposes. No operating limits or Surveillance Requirements are relaxed. The wording of current Technical Specifications SRs 3.4.9.1, 3.4.9.2, 3.4.9.5, 3.4.9.6, and 3.4.9.7 could result in overly conservative application of the requirements. The proposed amendment is written to remove the ambiguity in that the Applicability and Frequency of each Surveillance Requirement are clear. Neither the acceptance criteria nor the Surveillance Frequency of any Surveillance is reduced. Furthermore, the four possible RPV configurations are all adequately monitored. As a result, the margin of safety for the low pressure conditions is not significantly reduced due to the proposed changes.

The Unit 1 operating curves were extended to 32 EFPY using approved methodologies. More operational margin is provided, because the three vessel regions (upper vessel and flange, beltline, and bottom head) are being separated for the inservice leak and hydrostatic testing curve, and the heatup and cooldown curve. Although this separation results in more operating margin for certain vessel regions, it does not represent a significant reduction in the margin of safety. As described previously, the current Technical Specifications curves represent a composite of the three regions. Thus, the curves represent the temperature for the limiting region at a particular point. The regions that are not limiting at a particular point are artificially restricted. Separating the three regions, as proposed, eliminates false limits. The true limit for each region is preserved and uncompromised, based on the use of approved methodologies.

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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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NRC Project Director: Herbert N. Berkow

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 amendment request: October 7, 1996

Description of amendment request: The proposed changes to Plant Hatch Unit 1 and Unit 2 Technical Specifications (TS) would revise Surveillance Requirements (SR) 3.1.7.7 and 3.4.3.1, and Limiting Conditions for Operation (LCO) 3.4.3, 3.5.1, and 3.6.1.6, to increase the nominal mechanical pressure relief setpoints for all of the 11 safety/relief valves (SRV) to 1150 psig and allow operation with one SRV and its associated functions inoperable. The proposed changes would reduce the potential for SRV pilot leakage and the potential for forced outages due to an inoperable SRV during a fuel cycle.

The existing TS require that during continuous operation, all of the 11 SRVs remain OPERABLE in the safety mode, 7 in the Automatic Depressurization System (ADS) mode, and 4 in the Low-Low Set (LLS) mode. If one SRV is inoperable for longer than the duration specified in the applicable Action Statements, the plant must be placed in a Cold Shutdown Condition. Analyses have been completed which show that, with one SRV out of service, all transient/accident criteria can still be met. Increasing the nominal mechanical relief setpoints will increase the simmer margin (i.e., the difference between the SRV setpoints and the vessel steam dome pressure), thereby potentially reducing SRV pilot leakage which may occur during a typical operating cycle.

As a result of increasing the mechanical relief setpoints for the SRVs, the Standby Liquid Control (SLC) System pump test discharge pressure is increased to 1232 psig. The High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems are capable of operating at this increased pressure.

In support of the proposed changes, General Electric (GE) prepared NEDC-32041P, ''Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Updated Safety/Relief Valve Performance Requirements,'' Revision 2, dated April 1996, which was included in the submittal.

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

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

The SRVs serve to mitigate postulated transients and accidents; the proposed changes do not alter the function or mode of operation of the SRVs. The probability of an OPERABLE or an INOPERABLE SRV inadvertently opening or failing to open or close is not affected by these changes. Therefore, the probability of an accident is not increased. Analysis[(a)] has been performed which considers the consequences of the various transients and accidents with the increased setpoints and with one SRV inoperable. The analysis also considers the impact on ECCS [Emergency Core Cooling System] performance, including HPCI and RCIC. The analysis has shown that the consequences of an accident with the increased SRV setpoints and with one SRV inoperable are not increased.

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

Revising the nominal SRV setpoint only changes when the SRV opens in its mechanical relief mode; the operation of the SRV and any other existing equipment is not altered. Operation with one SRV inoperable was evaluated[(a)] and does not introduce any new failure modes. The impact on the operation and design of other systems and components has been evaluated,[(a)] including ECCS and SLC. No new operating modes or failure modes are introduced. Thus, these changes do not contribute to a new or different type of accident.

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

The change in SRV setpoint and operation with one SRV inoperable was evaluated relative to the applicable safety system settings and found to remain acceptable. For example, the proposed changes were evaluated against peak clad temperature limits, ECCS operation, ASME Code overpressurization limits, the MINIMUM CRITICAL POWER RATIO Safety Limit, and containment design limits; no significant reduction in the margin of safety was identified[(a)].

(a) GE Report NEDC-32041P, ''Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Updated Safety/Relief Valve Performance Requirements, Revision 2 (Proprietary), April 1996''.

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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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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 amendment request: October 29, 1996

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) for Plant Hatch Units 1 and 2 associated with the installation of a digital Power Range Neutron Monitoring (PRNM) system and the incorporation of long-term stability solution hardware.

In response to Generic Letter 94-02, ''Thermal-Hydraulic Instabilities in Boiling Water Reactors,'' Georgia Power Company (GPC) selected General Electric (GE) Option III as the long-term stability solution. Option III detects core instabilities and provides a reactor scram signal to the Reactor Protection System (RPS). The long-term stability solution, GE Option III, is supported by the BWR Owners' Group Topical Report NEDO-31960-A submitted to the NRC for approval in May 1991, and NEDO-31960-A, Supplement 1, submitted to the NRC for approval in March 1992. The NRC issued a Safety Evaluation Report (SER) for NEDO-31960-A and Supplement 1 in July 1993. BWR Owners' Group Topical Report NEDO-32465, submitted to the NRC in June 1995, provides additional analysis for the detection and suppression methodology (Option III).

To execute the stability solution software, the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) electronics would be replaced with a PRNM system based on digital GE Nuclear Measurements Analysis and Control NUMAC modules. Implementation of the PRNM would affect the RPS and Control Rod Block TS 3.3.1.1, 3.3.2.1, 3.4.1 and 3.10.8.

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

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

The purpose of the proposed amendment is to incorporate the Power Range Neutron Monitoring (PRNM) retrofit and Oscillation Power Range Monitor (OPRM) installation. The types of Average Power Range Monitor (APRM) Functions that are credited to mitigate accidents were previously evaluated. The proposed OPRM Upscale Function is implemented in the same hardware that implements the APRM Functions. The change to a two-out-of-four RPS [Reactor Protection System] logic was analyzed and determined to be equal to the original logic.

The modification involves equipment that is intended to detect the symptoms of some accidents and initiate mitigating action. The worst case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram), but no failure can cause an accident. As discussed in the bases for proposed changes, the PRNM replacement system is designed to perform the same operations as the existing Power Range Monitoring (PRM) system and to meet or exceed all of its operational requirements. Therefore, it is concluded that the probability of an accident previously evaluated is not increased as a result of replacing the existing equipment with the PRNM equipment.

* * * *

Human-machine interface (HMI) failures in the current system could be related to incorrectly adjusted settings, incorrect reading of meters, and failure to return the equipment to the normal operating configuration. There are comparable failure modes for some of these problems in the digital system where an erroneous potentiometer adjustment in the current system is equivalent to an erroneous digital entry in the replacement system. Certain potential ''failure to reconfigure errors'' in the current system have no counterpart in the replacement system, because any reconfiguration'' is automatically returned to normal by the system. Also, since parameters are available for review at any time, even if an error, such as a digital entry error occurs, it is more likely that the error would be almost immediately detected by recognition that the displayed value is not the correct one.

The failure analysis of the current system assumes certain rates of human error. The rates for the replacement system will be lower and, hence, are bounded by the FSAR [Final Safety Analysis Report] analysis.

Therefore, GPC [Georgia Power Company] concludes the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The APRM Trip Functions credited in the accident analyses are retained in the PRNM retrofit. The response time of the new electronics meets or exceeds the required response criteria. No new interfaces or interactions with other equipment will introduce any new failure modes.

The modification involves equipment that is intended to detect the symptoms of some accidents and initiate mitigating action. The worst-case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram), but no failure can cause an accident. This is unchanged from the current system.

Software common-cause failures can at most cause the system to fail to perform its safety function. In that case, it could fail to initiate action to mitigate the consequences of an accident, but would not cause one.

The new system is a digital system with software (firmware) control. As such, it has ''central'' processing points and software controlled digital processing where the current system had analog and discrete component processing. The result is that the specific failures of hardware and potentially common-cause software failures are different from the current system. Also, automatic self-test results in some cases in a direct trip as a result of a hardware failure where the current system may have remained ''as-is''. However, when these are evaluated at the system level, there are no new effects. In general, FSARs assume simplistic failure modes (relays for example) but do not specifically evaluate such effects as self-test detection and automatic trip or alarm.

The effects of software common-cause failure are mitigated by hardware design and system architecture. The replacement equipment is fully qualified to operate in its installed location and will not affect other equipment.

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

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

The replacement equipment provides the same function as the original electronics. Response time and operator information are either maintained or improved. The equipment was qualified, where appropriate, to assure its intended safety function is performed. The replacement system has improved channel trip accuracy compared to the current system and meets or exceeds system requirements assumed in setpoint analysis. The channel response time exceeds the requirements. The channel indicated accuracy is improved over the current system, and meets or exceeds system requirements. The replacement system meets or exceeds all system requirements.

The BWROG [BWR Owners' Group] Stability Option III was developed to meet the requirements of GDC [General Design Criterion] 10 and GDC 12 by providing a hardware system that detects the presence of thermal-hydraulic instabilities and automatically initiates the necessary actions to suppress the oscillations prior to violating the MCPR [maximum critical power ratio] Safety Limit. The NRC has reviewed and accepted the Option III methodology described in Licensing Topical Report NEDO-31960 and concluded this solution will provide the intended protection. Therefore, it is concluded that there will be no reduction in the margin of safety as defined in the Technical Specifications as a result of the installation of the OPRM system and the simultaneous removal of the operating restrictions imposed by the ICAs [item control areas].

Therefore, GPC concludes 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: November 20, 1996

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to allow the Vice President to designate the Safety Audit and Review Committee (SARC) Chairperson, to change the work hours limitation in accordance with guidance in GL 82-12, ''Nuclear Power Plant Staff Working Hours;'' to change radioactive shipments record retention requirements to comply with recent 10 CFR Part 20 changes; and other editorial changes.

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

1. The changes requested are administrative in nature. Paragraph 3.D was placed in the License by Amendment No. 155 to authorize Omaha Public Power District (OPPD) to increase the storage capacity of the FCS spent fuel pool. Amendment No. 155 stated that the TS as issued would be effective when the last new rack was installed. Since the last new rack was installed on August 8, 1994, Paragraph 3.D is no longer necessary and should be deleted from the License.

Table of Contents, Section 6.0, ''Interim Special Technical Specifications,'' Subsections 6.1 through 6.4 are proposed for deletion because all of the Specifications referred to have been deleted by previous Amendments.

The revision proposed for TS 2.15 (Item 2C of Table 2-3 & Item 1C of Table 2-4) will insert the correct terminology (Pressurizer Low/Low Pressure) into the Functional Unit description.

The revision proposed for TS 5.2 will require the control of overtime worked by personnel to be in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12) in lieu of stating the specific times requirements from the Policy as the current TS does. This option is in accordance with NUREG-1432, Standard TS for Combustion Engineering Plants, Specification 5.2.2e, and will allow work groups to be on twelve hour shifts.

The revision proposed for TS 5.5.2.2 will replace the specific title of the Chairperson of the Safety Audit and Review Committee and replace it with ''Member as appointed by the Vice President.'' This will allow the flexibility to change chairmanship of the committee amongst the members.

The revision to TS 5.10 concerning retention of records of radioactive shipments will update the TS to current 10 CFR 20 requirements. Plant procedures already comply with current 10 CFR 20 record retention requirements. The addition of the Section 5.0 title corrects a minor format discrepancy.

These proposed revisions are administrative in nature. The proposed revisions have no effect on any initial assumptions or operating restrictions assumed in any accident, nor do these changes have any effect on equipment required to mitigate the consequences of an accident. Therefore the proposed revisions 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 revisions correct minor errors, remove outdated information, are consistent with changes in organizational structure, 10 CFR Part 20, or NUREG-1432, ''Combustion Engineering Standard Technical Specifications (STS). These changes will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. No new operating modes are proposed as a result of these changes. 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 revisions listed above correct minor errors, remove outdated information, or are consistent with changes in organizational structure, 10 CFR Part 20, or Standard TS. These changes will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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: October 28, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.8.1, ''A.C. Sources,'' TS Section 3/4.8.2, ''Onsite Power Distribution Systems,'' TS Table 4.8.1, ''Battery Surveillance Requirements,'' and the associated bases. Surveillance requirements would be modified to account for the increase in the fuel cycle, consistent with Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,'' dated April 2, 1991. Administrative changes are also proposed.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the A.C. Offsite Sources, the Emergency Diesel Generators and the Station Batteries or the proposed revision to remove the ''during shutdown'' restriction for conduct of the battery performance test.

Results of the review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews.

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

Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

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

The proposed revision to reflect that the battery charger performance test will continue to be conducted on a[n] 18 month surveillance interval is an administrative change and does not affect previously analyzed accidents.

The proposed revision to the Bases to reflect that a change to a 24 month surveillance test interval is an exception to current guidance is an administrative change and does not affect previously analyzed accidents.

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

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

No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval and to restrictions on conducting testing only during shutdown conditions.

Results of the review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews.

The proposed revision to reflect that the battery charger performance test will continue to be conducted on a[n] 18 month surveillance interval is an administrative change and does not alter testing currently being performed.

The proposed revision to the Bases to reflect that a change to a 24 month surveillance test interval is an exception to current guidance is an administrative change and does not alter testing currently being performed.

3. Not involve a significant reduction in a margin of safety because the results of the historical 18 month surveillance data and maintenance records review identified no potential for a significant increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, consequently there are no significant reductions in a margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: November 26, 1996

Description of amendment request: The proposed changes would eliminate the records retention requirements from the administrative section of the Technical Specifications (TS) in accordance with NRC Administrative Letter95-06, ''Relocation of Technical Specifications Administrative Controls Related to Quality Assurance.''

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

Specifically, operation of the ... North Anna Power [Station] in accordance with the proposed Technical Specifications changes will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed administrative changes do not affect equipment or its operation. Therefore, the likelihood that an accident will occur is neither increased nor decreased by relocating record retention requirements from the Technical Specifications to the Operational Quality Assurance Program. This TS change will not impact the function or method of operation of plant equipment. Thus, a significant increase in the probability of a previously analyzed accident does not result due to this change. No systems, equipment, or components are affected by the proposed changes. Thus, the consequences of any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are not increased by this change.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not alter the design or operations of the physical plant. Since record retention requirements are administrative in nature, a change to these requirements does not contribute to accident initiation, an administrative change related to this activity does not produce a new accident scenario or produce a new type of equipment malfunction. [These] changes do not alter any existing accident scenarios. The proposed administrative change does not affect equipment or its operation, and, thus, does not create the possibility of a new or different kind of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

(3) Involve a significant reduction in a margin of safety. Section 6.0 of the North Anna ... Technical Specifications does not have a basis description. The proposed administrative change does not affect equipment or its operation, and, thus, does not involve any 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: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

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

NRC Project Director: Mark Reinhart, Acting

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

Date of amendment request: December 3, 1996

Description of amendment request: This amendment request proposes to revise the technical specifications associated with the inspection of the reactor coolant flywheel to provide an exception to the recommendations of Regulatory Guide 1.14, Revision 1, ''Reactor Coolant Pump Flywheel Integrity.'' The proposed exception would allow either an ultrasonic volumetric examination or surface examination to be performed at approximately 10-year intervals. In addition, a correction of the issuance date of a referenced regulatory guide is included.

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

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

The safety function of the RCP [reactor coolant pump] flywheels is to provide a coastdown period during which the RCPs would continue to provide reactor coolant flow to the reactor after loss of power to the RCPs. The maximum loading on the RCP flywheel results from overspeed following a LOCA [loss-of-coolant accident]. The maximum obtainable speed in the event of a LOCA was predicted to be less than 1500 rpm. Therefore, a peak LOCA speed of 1500 rpm is used in the evaluation of RCP flywheel integrity in WCAP-14535. This integrity evaluation shows a very high flaw tolerance for the flywheels. The proposed change does not affect that evaluation. Reduced coastdown times due to a single failed flywheel is bounded by the locked rotor analysis, therefore, it would not place the plant in an unanalyzed condition. Therefore, these 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 amendment does not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the proposed change, since the proposed change does not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

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

The operating limits and functional capabilities of the affected systems, structures, and components are basically unchanged by the proposed amendment. The results of the flywheel inspections performed have identified no indications affecting flywheel integrity. As identified in WCAP-14535, detailed stress analysis as well as risk analysis have been completed with the results indicating that there would be no change in the probability of failure for RCP flywheels if all inspections were eliminated. 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 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 3, 1996

Description of amendment request: This amendment request proposes to correct the reference to the Action Statement for Item 7.b, RWST Level - Low-Low Coincident with Safety Injection, Table 3.3-3, Engineered Safety Features Actuation System Instrumentation, from Action 16 to Action 28.

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.

Changing the reference from Action Statement 16 to Action Statement 28 for Functional Unit 7.b. of Table 3.3-3 will reduce the probability for an automatic switchover from the RWST [refueling water storage tank] to an empty containment sump to occur, while an RWST level channel is inoperable or is being tested with its bistable tripped, should an inadvertent safety injection signal occur concurrent with a single failure of a second RWST level channel. The design of these channels does not allow for operation or testing in bypass, so Action Statement 16 is not applicable. Changing to Action Statement 28 will limit the duration that a channel could be inoperable or be in test with its bistable bypassed. This change does not involve any design changes or hardware modifications, and does not introduce any new potential accident initiating conditions. The increase in allowed outage time for this item was evaluated and the associated unavailability and risk was shown to be equivalent to, or less than, that of other functional units evaluated in WCAP-10271, Supplement 2, Revision 1. Therefore, this proposed change does not increase the probability of any 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 result in any hardware changes and does not result in a change in the manner in which the ESFAS [engineered safety features actuation system] provides plant protection. This change does not alter the functioning of the ESFAS. Rather, the likelihood or probability of the ESFAS functioning properly is affected as described above. This change will not change the method by which any safety-related system performs its function. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

This proposed change will not result in a significant reduction in the margin of safety defined for any technical specification since it does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined.

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: June 28, 1996

Brief description of amendments: The amendment would modify the technical specifications (TS) to increase the minimum required amount of anhydrous trisodium phosphate (TSP) in the containment baskets. TSP is used to ensure that following a postulated design basis loss of coolant accident (LOCA), the containment sump pH is maintained greater than or equal to seven.

Date of issuance: December 10, 1996

Effective date: December 10, 1996, to be implemented within 45 days from the date of issuance.

Amendment Nos.: Unit 1 - 110; Unit 2 - 102; Unit 3 - 82

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

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47962) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 10, 1996.No significant hazards consideration comments received: No.

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

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: June 21, 1996

Brief description of amendments: The amendments revise the term ''lifting loads'' used in Technical Specification 3.9.6b.2, Manipulator Crane, to ''lifting force.'' This revision will clarify that the static loads associated with the lifting tool, drive rod, and control rod weights are not included in the lifting force limit.

Date of issuance: December 12, 1996

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

Amendment Nos.: 171 and 153

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

Date of initial notice in Federal Register: September 11, 1996 (61 FR 47977) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 22, 1996, and as supplemented by letters dated July 4 and September 20, 1996

Brief description of amendment: The amendment revises Clinton Power Station Technical Specification 3.3.4.1, ''End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation,'' by deleting Surveillance Requirement 3.3.4.1.6 which requires the RPT breaker interruption time to be determined at least once per 60 months.

Date of issuance: December 13, 1996

Effective date: December 13, 1996

Amendment No.: 111

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18169) The supplemental letters of July 4 and September 20, 1996, provided clarifying information and did not include significant changes relative to the original Federal Register notice.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 13, 1996.No significant hazards consideration comments received: No

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

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: July 12, 1996, as supplementedOctober 30, 1996.

Brief description of amendment: The amendment revises TS 6.2.2.h regarding the administrative controls for the normal working hours of unit staff who perform safety-related functions, and TS 6.2.2.i regarding an organizational change. The changes authorize (1) establishment of unit staff work schedules that average 40 hours per week using shifts as long as 12 hours, and (2) elimination of the positions of General Supervisor Operations and Supervisor Operations.

Date of issuance: December 12, 1996

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

Amendment No.: 158

Facility Operating License No. DPR-63: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42280) The October 30, 1996, letter provided supplemental information that did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1996.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.

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: July 12, 1996

Brief description of amendment: The amendment revises Technical Specification Section 6.2.2.i regarding the administrative controls for the normal working hours of unit staff who perform safety-related functions. The change allows the establishment of unit staff work schedules that average 40 hours per week using shifts as long as 12 hours.

Date of issuance: December 12, 1996

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

Amendment No.: 78

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42281) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1996.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.

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: July 28, 1995, as supplemented October 25, 1995, and August 9, 1996

Brief description of amendments: The amendments revise the 250 volt DC profiles in the Technical Specifications for the two units to reflect new load profile calculations.

Date of issuance: December 17, 1996

Effective date: Unit 1, as of date of issuance, to be implemented within 30 days; Unit 2, as of date of issuance, to be implemented prior to Startup following the Eighth Refueling and Inspection Outage for Unit 2, which is scheduled for the Spring of 1997.

Amendment Nos.: 162 and 133

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

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47622) The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the Federal Register notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 12, 1992, as supplemented September 17, 1992, March 17, 1993, August 17, 1993, August 18, 1993, December 29, 1993, June 29, 1995, August 15, 1996, October 3, 1996,October 23, 1996, November 14, 1996, November 20, 1996 (JPN-96-045), November 20, 1996 (JPN-96-046), and November 27, 1996.

Brief description of amendment: The amendment modifies

Facility Operating License No. DPR-59 and the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Technical Specifications (TSs) to authorize an increase in the maximum power level of JAFNPP from 2436 MWt to 2536 MWt. The amendment also approves changes to the TSs to implement uprated power operation.

Date of issuance: December 6, 1996

Effective date:

As of the date of issuance to be implemented upon plant startup following the refueling outage cycle 13.

Amendment No.: 239

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

Date of initial notice in Federal Register: February 2, 1994 (59 FR 4943) The letters dated September 17, 1992, March 17, 1993, August 17, 1993, August 18, 1993, December 29, 1993, June 29, 1995, August 15, 1996,October 3, 1996, October 23, 1996, November 14, 1996, November 20, 1996, (JPN-96-045), November 20, 1996, (JPN-96-046), and November 27, 1996, 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 6, 1996.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.

Public Service Electric & Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey Date of application for amendment: September 20, 1996, as supplemented September 30, 1996

Brief description of amendment: The amendment changes Technical Specification Surveillance Requirement 4.7.7.b.4 for the Auxiliary Building Exhaust Air Filtration System, and its associated Bases, to indicate that the specified flowrate applies only to system testing.

Date of issuance: December 12, 1996

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

Amendment No. 168

Facility Operating License No. DPR-75: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55040) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 27, 1996, as supplemented October 24, 1996

Brief description of amendments: The amendment to Unit 2 deletes License Condition 2.C.(24)(a) which required establishment by June 3, 1981, of regularly scheduled 8-hour shifts without reliance on routine use of overtime. The amendments to both Units 1 and 2 revise Technical Specification 6.2.2 to delete the reference to Generic Letter 82-12, ''Nuclear Plant Staff Working Hours,'' and require that administrative controls be established which will ensure that adequate shift coverage is maintained without heavy use of overtime for individuals.

Date of issuance: December 17, 1996

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

Amendment Nos. 186 and 169

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications for both units and License for Unit 2 only.

Date of initial notice in Federal Register: September 12, 1996 (61 FR 48175) The October 24, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or the original notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: May 29, 1996

Brief description of amendments: These amendments revise Technical Specification (TS) Surveillance Requirement 3.5.1.4 to increase the minimum boron concentration in the safety injections tanks from 1850 ppm to 2200 ppm.

Date of issuance: December 6, 1996

Effective date: December 6, 1996, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2 - 135; Unit 3 - 124

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

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40029) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 6, 1996. No significant hazards consideration comments received: No.Temporary

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

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

Date of application for amendment: September 27, 1996, as supplemented on October 25, and November 18, 1996

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant Technical Specification requirements related to the low temperature overpressure protection (LTOP) system. Specifically, the LTOP curve is modified to define 10 CFR Part 50, Appendix G pressure temperature limitations for LTOP evaluation through the end of operating cycle (EOC) 33. In addition, the LTOP enabling temperature and the temperature required for starting a reactor coolant pump have been changed consistent with the design basis for the LTOP system. Finally, the TS bases were changed consistent with the changes described above.

Date of issuance: December 13, 1996

Effective date: December 13, 1996, to be implemented within 30 days.

Amendment No.: 130

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

Date of initial notice in Federal Register: October 7, 1996 (61 FR 52472) The October 25 and November 18, 1996, submittals provided supplemental 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 13, 1996.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

Dated at Rockville, Maryland, this 24th day of December 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 96-33254 Filed 12-31-96; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–38086; File No. SR–CBOE–96–69]

Self-Regulatory Organizations; Notice of Filing and Order Granting Accelerated Approval of Proposed Rule Change by the Chicago Board Options Exchange, Incorporated Relating to Calculating Blue Sheets Violation Aggregate Fines on a Rolling Year Basis

December 26, 1996.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1], and Rule 19b–4 [2] thereunder, notice is hereby given that on November 20, 1996,[3] the Chicago Board Options Exchange, Incorporated (''CBOE'' or ''Exchange'') filed with the Securities and Exchange Commission (''SEC'' or

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] On December 17, 1996, the Exchange filed Amendment No. 1 to the proposed rule change. Amendment No. 1 is a technical amendment, correcting Exhibit I, Section I to the filing. *See* letter from Margaret Abrams, Senior Attorney, CBOE to Janice Mitnick, Attorney, Division of Market Regulation, SEC, dated December 17, 1996.