NUCLEAR REGULATORY COMMISSION

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 November 22, 1996, through December 6, 1996. The last biweekly notice was published on December 4, 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 January 17, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of application for amendment request: September 20, 1996.

Description of amendment request: The proposed amendments would update the Pressure Temperature (P–T) curves contained in the Technical Specifications to 22 Effective Full Power Years (EFPYs).

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

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

The proposed changes merely adjust the reference temperature for the limiting beltline material to account for irradiation effects and provide the same level of protection as previously evaluated. The adjusted reference temperature calculations were performed utilizing the guidance contained in Regulatory Guide 1.99, Revision 2. The change is administrative in nature to reflect the extension of the operating limits to 22 EFPY. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

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

The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. The revised operating limits are merely an update of the old limits by taking into account the effects of irradiation on the limiting reactor vessel material. Use of the revised P–T curves will continue to provide the same level of protection as was previously reviewed and approved. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated change to the P–T curves related to this proposed amendment does not affect any activities or equipment and are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment reflects an update of the P–T curves to extend the operating limit to 22 EFPY. The revised curves are based on the latest NRC guidance along with actual data for the units. The new limits retain the margin of safety to the level expected for a new vessel, adjusted for irradiation effects as required by 10 CFR, Appendix G, thereby maintaining a conservative margin of safety.

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

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: October 31, 1996.

Description of amendment request: The proposed amendments would relocate the requirements for seismic monitoring instrumentation from the Technical Specifications to licensee controlled documents. The Technical Specifications affected are 3/4.3.7.2, ''Seismic Monitoring Instrumentation,'' Table 3.3.7.2–1, ''Seismic Monitoring Instrumentation,'' Table 4.3.7.2–1, ''Seismic Monitoring Instrumentation Surveillance Requirements,'' and Bases Section 3/4.3.7.2, ''Seismic Monitoring Instrumentation.''

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

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

The function of the seismic monitoring instrumentation is to monitor seismic activity above the Operating-Basis Earthquake (OBE) threshold, and to record observed seismic data for comparison to design basis response spectra. The seismic monitoring instrumentation does not provide any function to mitigate an accident or the consequences of an accident. The replacement seismic monitoring instrumentation will remain in place. The proposed Amendment is not a result of any changes to system function, alarm setpoints, or main control room annunciators. Rather, the Technical Specification requirements (as revised for the replacement instrumentation) are being relocated to licensee-controlled documents in accordance with NRC Generic Letter 95–10.

The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the LaSalle Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The seismic monitoring instrumentation does not provide any function to mitigate an accident or the consequences of an accident. The replacement seismic monitoring instrumentation will remain in place and will provide the same basic function as the existing instrumentation. The replacement instrumentation will provide enhanced system reliability and will not result in any changes to system function, alarm setpoints, or main control room annunciators. The Technical Specification requirements (as revised for the replacement instrumentation) are being relocated to licensee-controlled documents in accordance with NRC Generic Letter 95–10.

The proposed change does not involve any change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The replacement seismic monitoring instrumentation will have no impact on margin of safety. The intended function of the seismic monitoring instrumentation, i.e. to record observed seismic data for analysis to determine the impact on plant components, will be made more reliable by this modification. The Technical Specification requirements (as revised for the replacement instrumentation) are being relocated to licensee-controlled documents in accordance with NRC Generic Letter 95–10.

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable continue to meet the same requirements as the existing Technical Specifications. However, the LCO requirement specified in Section 3.3.7.2.a (to prepare and submit a Special Report to the NRC within 10 days of the seismic monitoring instrumentation being inoperable for more than 30 days) will not be included in the ATR [Administrative Technical Requirements] since the Technical Specification Special Report requirements are only applicable to the LCOs. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specification, NUREG–1434, Rev. 1 approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: November 7, 1996.

Description of amendment request: The proposed amendments would change Specification 4.3.1.A.4.b from verifying greater than or equal to 17 percent steam generator secondary side wide range water level to greater than or equal to 17 percent steam generator secondary side narrow range water level.

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 occurrence of any accident previously evaluated.

Maintaining secondary side steam generator water level greater than or equal to 17 percent by wide range level indication is the current requirement by the technical specifications. By revising the requirement to require using the narrow range water level, no change in operating practices or plant configuration is made. The minimum requirement of 17 percent by narrow range level indication is more restrictive and conservative than 17 percent by wide range indication. The requirement to maintain secondary side steam generator water level greater than or equal to 17 percent by narrow range indication is currently required by operations procedure PT-O, Appendix F–1 and will be maintained. This change ensures that the requirements for natural circulation cooldown are maintained in Mode 4. Therefore, changing this surveillance requirement does 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 proposed changes do not require a physical alteration of the plant (no new or different equipment will be installed). The Technical Specifications will continue to require OPERABLE steam generator(s) for heat removal functions. The Technical Specifications will continue to require the performance of SR 4.3.1.A.4.b. Changing the SR to use narrow level indication correctly states the steam generator water level required to support heat removal function. Thus, this change does 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 proposed changes do not result in a significant reduction in a margin of safety because it has no impact on any safety analysis assumptions. The requirement to have OPERABLE steam generator(s) in MODE 4 for heat removal function is maintained. The requirement to perform SR 4.3.1.A.4.b is not changed. Changing the SR to use narrow level indication correctly states the steam generator water level required to support heat removal function. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: November 7, 1996.

Description of amendment request: The proposed amendments would change the values for the reduced power range neutron flux high setpoint trip that are specified when one or more code main steam safety valves are inoperable.

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 occurrence of any accident previously evaluated.

The requirement to change the Power Range Neutron Flux High Trip setpoints to the Reduced Setpoint Values of Table 3.7–1 for the most restrictive loop if one or more code MSSVs are inoperable is not changed by this amendment. As such, no change in operating practices or plant configuration is being made.

The amendment provides new reduced setpoint values for the Power Range Neutron flux High Trip to ensure that for the limiting transient (Loss of Load/Turbine Trip [LOL/TT]), a secondary side overpressurization condition does not occur. The new values were the result of calculation using an algorithm provided by Westinghouse in Westinghouse Nuclear Safety Advisory Letter NSAL-94–001, ''Operation at Reduced Power Levels with Inoperable MSSVs,'' January 25, 1994. The new values are much more restrictive than the previous values and ensure that the probability or consequences of an accident previously evaluated is not increased. Therefore, the new reduced setpoint values 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 proposed change does not require a physical alteration of the plant (no new or different equipment will be installed to implement this change). The Reduced Neutron Flux High Trip setpoints ensure that a secondary side overpressurization transient does not occur for the most limiting transient. In addition, no new modes of operations will be introduced by this change. Thus, this change does 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.

This amendment provides new Reduced Power Range Neutron Flux High Trip setpoints.The Specification that requires the Power Range Neutron Flux High Trip setpoints be changed to the reduced values for one or more inoperable MSSVs is not changed. The reduced Trip setpoints are the result of new calculations using an algorithm provided by Westinghouse in Westinghouse Nuclear Safety Advisory Letter NSAL–94–001, ''Operation at Reduced Power levels with Inoperable MSSVs,'' January 25, 1994, and ensure the LOL/TT transient does not result in a secondary overpressurization. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: November 7, 1996.

Description of amendment request: The proposed amendments would clarify the operability requirements for the residual heat removal (RHR) loops during core alteration operations.

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 occurrence of any accident previously evaluated.

The ability to remove an RHR loop from operation for up to one hour per eight-hour period is currently allowed by technical specification 3.13.9.B.b. By adding a reference to LCO [Limiting Condition for Operation] 3.13.1.A.4. and adding the requirement to suspend CORE ALTERATIONS to Action 3.13.9.B.a. to be consistent with 3.13.9.B.b., no change in operating practices or plant configuration is made. By maintaining the requirement to have an RHR loop in operation during MODE 6, and by requiring CORE ALTERATIONS to be suspended if an RHR loop is not back in operation after one hour, adequate corrective actions are implemented until the RHR loop is restored to operating status. Therefore, operation of the system is consistent with current technical specifications and this change does 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 proposed changes do not require a physical alteration of the plant (no new or different equipment will be installed to implement this change). The Technical Specifications will continue to require an RHR loop to be in operation during MODE 6, and will only permit the loop to be not in operation for up to one hour in an eight-hour period. The Technical Specifications will continue to require compliance with these limitations and suspension of CORE ALTERATIONS if an RHR loop is not in operation for more than one hour. Thus, this change does 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 proposed changes do not result in a significant reduction in a margin of safety because it has no impact on any safety analysis assumptions. The requirement to have an RHR loop in operation during MODE 6 is maintained, along with the ability to remove RHR from operation for up to one hour per eight-hour period. If an RHR loop is not in service beyond 1 hour per TS 3.13.9.B, CORE ALTERATIONS will be suspended. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One

First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

Consumers Power Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: November 7, 1996.

Description of amendment request: The proposed amendment would revise Technical Specification 4.2.9, Service and Instrument Air System, to add an additional air compressor.

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

The proposed change does not:

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

Utilizing the existing piping configuration, both the new and the existing air compressors are capable of supporting either portion of the Service and Instrument Air System. The addition of the fourth air compressor will decrease the probability of an accident previously evaluated, because capacity is being added to the system. The consequences of an accident previously evaluated will not be affected by the addition of a fourth air compressor. The Service and Instrument Air System performs the non-safety related function of providing compressed air for service use and moisture free compressed instrument air for control air demands. The instrument air portion is designed so that its operation is required for plant reliability, not plant nuclear safety. Safety-related equipment supplied by instrument air is designed to fail in its safe condition upon loss of instrument air or, safety-related equipment (and nonsafety-related equipment determined to be important to safety) required to operate subsequent to instrument air failure is supplied by backup nitrogen accumulators.

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

The operation of the equipment in the Service and Instrument Air System is essentially unchanged. The new air compressor is a similar design (nonlubricated), providing additional air volume at a quality comparable to the three existing air compressors. Therefore, the possibility of an accident of a different kind than any previously evaluated has not been created.

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

The Technical Specification does not specify a margin of safety for the operation of the Service and Instrument Air System, other than specifying that [''Instrument and service] air shall be supplied by three, nonlubricated air compressors, each rated at 70 scfm [standard cubic feet per minute]. Instrument air shall also pass through a dryer.'' Addition of a fourth air compressor will increase the available capacity, thus increasing the margin of safety. Therefore, adding the statement ''and one, nonlubricated air compressor rated at 100 scfm'' to Technical Specification 4.2.9. will not reduce the margin of safety.

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

Local Public Document Room location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: October 4, 1996.

Description of amendment request: The proposed amendment would revise the surveillance requirements in Technical Specifications (TSs) 4.1.2.3.1, 4.1.2.4.1, 4.5.2.b, and 4.6.2.1.b and associated Bases. The subject surveillance requirements are applicable to the charging/high head safety injection pumps, low head safety injection pump, and the containment quench spray pumps. The proposed changes would replace the current specific test acceptance criteria contained in these surveillance requirements with requirements to verify pump performance in accordance with the Inservice Testing Program, the Emergency Core Cooling System Flow Analysis, or the Containment Integrity Safety Analysis, as applicable. The proposed changes would also make minor editorial changes in these TSs and make conforming changes in the TS Index pages.

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?

The change does not result in a modification to plant equipment nor does it affect the manner in which the plant is operated. Since the physical plant equipment and operating practices are not changed, as noted above, there is no change in the probability of an accident previously evaluated.

The proposed change will not lower the pump performance operability criteria for the charging/high head safety injection, low head safety injection and quench spray pumps, as assumed in the safety analysis. The required values for developed pump head and flow will continue to satisfy accident mitigation requirements and will be maintained and controlled in the Inservice Testing (IST) Programs(s).

Since the proposed change does not lower the pump's performance acceptance criteria, as assumed in the safety analysis, the containment depressurization system will continue to meet its design basis requirements. The proposed change will not impose additional challenges to the containment structure in terms of peak pressure. The calculated offsite dose consequences of a design basis accident (DBA) will remain unchanged since the one hour release duration and source term remain unchanged. The ability of the emergency core cooling system (ECCS) subsystems to provide sufficient emergency core cooling capability in the event of a loss of coolant accident (LOCA) remains unchanged. Therefore, peak cladding temperatures during a LOCA will continue to remain within acceptable limits. The ability of the ECCS subsystems to provide sufficient long term core cooling capability in the recirculation mode during the accident recovery period remains unchanged. The charging pumps, as part of the boron injection system, will continue to provide sufficient flow to ensure negative reactivity control during each mode of facility operation. Future changes to the pump head and flow requirements will be made under the 10 CFR 50.59 process to ensure that the system performance requirements continue to be met.

The proposed change to the Bases section will ensure that safety analyses assumptions for assumed pump performance continue to be met. The words ''required developed head'' will be clearly defined to reflect that they refer to the value(s) assumed in the safety analysis for the pump's developed head at a specific or a given point. The proposed changes to the Index pages and the footnote in LCO 3.1.2.4 are administrative in nature and do not affect plant safety.

Based on the above discussion, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the method of operating the plant. The charging pumps will continue to be in service during plant operation and be available to perform their function as high head safety injection pumps. This proposed change does not pose additional challenges to the design or function of the charging pumps. The low head safety injection and quench spray systems are accident mitigation systems and are normally in standby. System operation would be initiated as required to mitigate the consequences of a DBA. The charging/high head safety injection, low head safety injection and quench pumps will continue to provide sufficient flow to mitigate the consequences of a DBA. These systems' operation continues [sic] [continue] to fulfill the safety functions for which they were designed and no changes to plant equipment will occur. As a result, an accident which is new or different than any already evaluated in the Updated Final Safety Analysis Report will not be created due to this change.

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

3. Does the change involve a significant reduction in a margin of safety? The surveillance requirements for demonstrating that the pumps are operable will continue to assure the ability of the system to satisfy its design function. Therefore, the proposed change will not affect the ability of these systems to perform their safety function.

The containment systems' design requirements to restore the containment to subatmospheric condition within one hour will continue to be satisfied. This proposed change does not have an effect on the containment peak pressure since the charging/high head safety injection, low head safety injection and quench spray pumps' performance requirements are not being lowered. The ability of the ECCS subsystems to provide sufficient emergency core cooling capability in the event of a LOCA remains unchanged. Therefore, peak cladding temperatures during a LOCA will continue to remain within acceptable limits. The ability of the ECCS subsystems to provide sufficient long term core cooling capability in the recirculation mode during the accident recovery period remains unchanged. The charging pumps, as part of the boron injection system, will continue to provide sufficient flow to ensure negative reactivity control during each mode of facility operation. There is no resultant change in dose consequences since source term remains unchanged and the containment will continue to reach a subatmospheric pressure within the first hour following a DBA.

Each pump's performance requirements will continue to be controlled in a manner to ensure safety analysis assumptions are met.

Therefore, based on the above discussions, it can be concluded that the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: October 31, 1996.

Description of amendment request: The proposed amendments will revise administrative controls Technical Specification (TS) 6.5.1, ''Facility Review Group (FRG),'' and TS 6.8, ''Procedures and Programs.'' The revisions to TS 6.5.1 reduce the scope of procedures and procedure changes which require review by the FRG, transfer approval of certain procedures from the Plant Manager to the FRG, and require copies of FRG meeting minutes be provided to the Plant Manager. The changes to TS 6.8 reflect the corresponding changes in TS 6.5.1, and expand the scope of the section on temporary changes to procedures.

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

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

The proposed amendments revise certain administrative controls involved with the on-site programmatic process for review and approval of plant procedures. Specifications that are in place to provide assurance that the unit operating staff qualifications are acceptable, and that written procedures are established, implemented and maintained for safety related activities are not being changed. The revisions are consistent with industry standards established pursuant to 10 CFR Part 50, Appendix B, and do not alter any parameter or equipment performance assumptions that are contained in plant safety analyses to evaluate the initiation or consequences of an accident. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendments will not change the physical plant or the modes of plant operation defined in the Facility License for either St. Lucie unit. Changes proposed for the administrative controls do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments revise certain administrative controls involving the on-site programmatic process for review and approval of plant procedures. The scope, or the requirement to establish, maintain, and implement procedures for activities that could affect nuclear safety are not being changed. The proposed changes are consistent with approved industry standards and do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: October 31, 1996 (TSCR 205).

Description of amendment request: The proposed change requests deletion of Technical Specification Table 3.5.2 which lists automatic primary containment isolation valves. In addition, this change request clarifies the applicability of an action statement which applies to several limiting conditions for operation in Section 3.5 and deletes closure time requirements for several automatic isolation valves in Section 4.5.F.

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 deletion of the automatic primary containment isolation valve Table 3.5.2 and closure times for several valves in Specification 4.5.F.1 are administrative in nature and do not affect the purpose, function, operability and testing requirements of the automatic primary containment isolation valves or the isolation condenser isolation valves. The required action contained in Specification 3.5.A.7 has been moved to the associated specifications and has not changed. Capitalizing definitions and deleting unneeded pages are also administrative changes which enhance the usability of the Technical Specifications. Therefore, the proposed changes do not increase the probability of occurrence or consequence of an accident previously evaluated.

2. The proposed changes are administrative and do not involve a physical change to plant configuration nor do they affect the performance of any equipment. Existing limiting conditions for operation and surveillance requirements are retained. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Deleting the list of valves in Table 3.5.2 and valve closure times in Specification 4.5.F.1 are administrative changes which do not affect the purpose or function of the automatic primary containment isolation valves. The listing of the automatic primary containment isolation valves and stroke time requirements will be in controlled plant procedures. Changes to the list or closure times can be made in accordance with review procedures required by Section 6.5 of the Technical Specifications and 10 CFR 50.59. Similarly, inserting the statement of required action in Specification 3.5.A.7 into the Specifications to which it applies does not modify the condition or the action to be taken and is an administrative change which clarifies the Technical Specifications. Therefore, the margin of safety is not reduced.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: November 12, 1996, as supplemented November 27, 1996 (TSCR 224).

Description of amendment request: The proposed technical specification change will reflect the implementation of the revised 10 CFR Part 20, ''Standards for Protection Against Radiation.''

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

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

The proposed revisions to the liquid release rate limits and bases and gaseous effluent bases will not result in a change in the types or amounts of effluents released nor will there be an increase in individual or cumulative radiation exposures. In addition, these changes do not impact the operation or design of any plant structures, systems, or components. These changes ensure compliance with 10 CFR 50.36a and 10 CFR 50 Appendix I and result in levels of radioactive materials in effluents being maintained ALARA [as low as is reasonably achievable]. The revision to the high radiation area controls and dose measurement distance will ensure areas are conservatively posted as high radiation areas in compliance with 10 CFR 20.1601(a)(1) and provide controls to ensure individuals are not overexposed. Other proposed changes consist of revisions to 10 CFR 20 references to recognize the new section numbers, and administrative controls for record keeping to maintain compliance with the new Part 20.

These changes will not result in a change to plant design or operation. Therefore, it can be concluded that the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not affect the plant design or operation nor do they result in a change to the configuration of any equipment. There will be no change in the types or increase in the amount of effluents released offsite.

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

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

The proposed revisions do not involve any change in the types or increase in the amount of effluents released offsite. The proposed changes do not involve any actual change in the methodology used in the control of radioactive wastes or radiological environmental monitoring. The methodology that will be used in the control of radioactive effluents and calculation of effluent monitor setpoints will result in the same effluent release rate as the current methodology now being used. The operational flexibility needed for releases allows the use of limits as proposed. In addition, the changes in measurement distances for determination of high radiation areas will not result in an increase in individual or cumulative occupational radiation exposures since it will result in a more conservative identification of high radiation areas. Compliance with the limits of the new 10 CFR 20.1301 will be demonstrated by operating within the limits of 10 CFR 50 Appendix I and 40 CFR 190. Thus, operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

GPU Nuclear Corporation, et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: August 29, 1996, as supplemented October 3, 1996. The October 3, 1996, submittal contained editorial changes only and did not change the initial no significant hazards consideration evaluation.

Description of amendment request: The purpose of this amendment request is to incorporate certain improvements from the Standard Technical Specifications for B&W Plants, NUREG–1430.

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

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the

consequences of an accident previously evaluated. The proposed amendment deletes limiting conditions for operation (LCOs) from the TMI–1 Technical Specifications that are no longer required to be addressed in Technical Specifications per 10 CFR 50.36(c)(2)(ii). The proposed amendment deletes Surveillance Requirements from the TMI–1 Technical Specifications that are related to the LCOs to be deleted. These items are addressed in licensee controlled documents. Certain design feature specifications are also to be deleted consistent with the RSTS [Revised Standard Technical Specifications] for B&W plants. The proposed changes do not modify the operation, limits or controls of systems, structures or components relied upon to prevent or mitigate the consequences or accidents previously evaluated.

Also, the reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Therefore, this change does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

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

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: September 20, 1996.

Description of amendment request: The proposed amendment would revise the Nine Mile Point Unit 1 (NMP1) Technical Specifications that involve the frequencies of surveillance requirements stated in Tables 4.6.2a, 4.6.2b, 4.6.2g, and 4.6.11, and Sections 4.2.5b(1), 4.3.2b, 4.3.6b(1), 4.3.6b(2), 4.3.6b(3), 4.3.6b(4), 4.3.6c(2), 4.6.13b.1, and 4.6.13b.2. The surveillances associated with these tables and sections are currently satisfied during NMP1 refueling outages prior to restart of the unit. The proposed changes would permit surveillance testing either while the reactor is operating or during outage periods not associated with refueling. The requirements of the surveillance sections and tables addressed by this request that are not changed to be performed at power are being changed to allow surveillance credit to be taken for performance of the associated surveillances while the plant is in the Cold Shutdown, Refueling, or Major Maintenance modes. In addition to these proposed changes, typographical errors are corrected.

Basis for proposed no significant hazards consideration determination: The licensee states that: ''The periods between surveillances will not be inappropriately lengthened. For the affected surveillances, NMP1 administrative controls will require that the interval between surveillance testing not exceed a period equal to 1.25 times the nominal 24 months frequency (no longer than 30 months). The NMP1 plant preventive maintenance and surveillance database will be revised accordingly.''

The licensee groups the systems affected by this request into four categories:

Category 1: The associated system will remain operable and able to automatically perform its safety function during performance of surveillances that satisfy the proposed surveillance requirement.

Category 2: The system is required for monitoring purposes only and provides no automatic safety actuation function and redundant, or redundant and alternate channels are available for required monitoring.

Category 3: There is no change in the system configuration or plant operating conditions during the performance of associated surveillances whether the plant is shutdown for refueling or shutdown for maintenance. The surveillances performed to meet the requirements of NMP1 Technical Specifications Tables 4.6.2a Parameter 8 and 4.6.2g Parameter 6 are included in this category and may also be completed in concurrence with a unit shutdown. The only difference between the proposed changes and the normal unit shutdown sequence is that the mode switch may be taken to ''Shutdown'' in order to scram the plant. The response of the plant is the same as it is under the current plant shutdown procedures. There are no other differences in testing techniques or testing criteria from those previously required by the NMP1 Technical Specifications.

Category 4: The system or equipment is isolated or out of service during the performance of the required surveillances. The associated surveillance may be performed concurrently with quarterly valve stroking, at which time the system or equipment is already out of service.

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 operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequence of an accident previously evaluated.

Each of the four categories [* * *] are evaluated separately below:

Category 1: The associated systems will remain operable and able to automatically fulfill as designed any required safety functions that may become necessary during performance of required surveillances. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No plant transients will be initiated as a result of the proposed changes. No initiator of any accident previously evaluated is adversely affected. No system required to actuate to respond to any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report] is adversely affected by the proposed change.

Category 2: The associated systems will be required for monitoring purposes only and provide no automatic safety actuation function and redundant, or redundant and alternate channels are available for required monitoring. Since redundant monitoring instrumentation will still be available as required by the technical specifications, the associated systems' functions in accident mitigation are not affected. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No plant transients will be initiated as a result of the proposed changes. No initiator of any accident previously evaluated is adversely affected. No system required to actuate to respond to any accident previously evaluated in the UFSAR is adversely affected by the proposed changes.

Category 3: There will be no change in the system configuration or plant operating conditions during the performance of associated surveillances. The associated system's ability to perform required safety functions will not be affected, whether the plant is shutdown for refueling or shutdown for maintenance. The surveillances performed to meet the requirements of NMP1 Technical Specifications Tables 4.6.2a Parameter B and 4.6.2g Parameter 6 are included in this category and may also be performed in concurrence with a unit shutdown. The only difference between the proposed changes and the normal unit shutdown sequence is that the mode switch may be taken to ''Shutdown'' in order to scram the plant. The response of the plant is the same as it is under the current plant shutdown procedures. There are no other differences in testing techniques or testing criteria from those previously required by the NMP1 Technical Specifications. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No unexpected plant transients will be initiated as a result of the proposed changes. No initiator of any accident previously evaluated is adversely affected. No system required to actuate to respond to any accident previously evaluated in the UFSAR is adversely affected by the proposed changes.

Category 4: The associated system or equipment will be isolated or out of service during the performance of the required surveillances. The associated surveillances will be performed during quarterly valve stroking, at which time the system or equipment is already out of service. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No plant transients will be initiated as a result of the proposed changes. No initiator of any accident previously evaluated is adversely affected. No system required to actuate to respond to any accident previously evaluated in the UFSAR is adversely affected by the proposed changes.

The correction of the typographical errors is administrative only and has no affect on plant systems or procedures. In all cases, equipment used for accident mitigation is not adversely affected. The ability of the operators to safely shut down NMP1 is not impaired. The changes will not adversely affect any accident precursor or initiator of any accident. For these reasons, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Each of the four categories [* * *] are evaluated separately below.

Category 1: The associated systems will remain operable and able to automatically perform as designed any required safety functions that may become necessary during performance of required surveillances. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No accident initiator or failure of a different type than previously identified in the UFSAR is introduced. No different or new plant transients may result from those previously evaluated in the UFSAR.

Category 2: The associated systems will be required for monitoring purposes only and provide no automatic safety actuation function. Since redundant, or redundant and alternate monitoring instrumentation will still be available as required by the technical specifications, the associated systems' functions in accident mitigation are not affected. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No accident initiator or failure of a different type than previously identified in the UFSAR is introduced. No different or new plant transients may result from those previously evaluated in the UFSAR.

Category 3: There will be no change in the system configuration or plant operating conditions during the performance of associated surveillances. The associated system's ability to perform required safety functions will not be affected, whether the plant is shutdown for refueling or shutdown for maintenance. The surveillances performed to meet the requirements of NMP1 Technical Specifications Tables 4.6.2a Parameter 8 and 4.6.2g Parameter 6 are included in this category and may also be performed in concurrence with a unit shutdown. The only difference between the proposed changes and the normal unit shutdown sequence is that the mode switch may be taken to ''Shutdown'' in order to scram the plant. The response of the plant is the same as it is under the current plant shutdown procedures. There are no other differences in testing techniques or testing criteria from those previously required by the NMP1 Technical Specifications. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No unexpected plant transients will be initiated as a result of the proposed changes. No accident initiator or failure of a different type than previously identified in the UFSAR is introduced. No different or new plant transients may result from those previously evaluated in the UFSAR.

Category 4: The associated system or equipment will be isolated or out of service during the performance of the required surveillance. The associated surveillances will be performed during quarterly valve stroking, at which time the system or equipment is already out of service. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No plant transients will be initiated as a result of the proposed changes. No accident initiator or failure of a different type than previously identified in the UFSAR is introduced. No different or new plant transients may result from those previously evaluated in the UFSAR.

The correction of the typographical errors is administrative only and has no affect on plant systems or procedures. In all cases, the changes will not adversely affect any accident precursor or initiator of any accident and, therefore, the changes do not introduce any new failure modes or conditions that may create a new or different accident. For these reasons, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

Each of the four categories [* * *] are evaluated separately below.

Category 1: The associated systems will remain operable and able to automatically perform required safety functions during performance of surveillances that satisfy the surveillance requirement. There will be no effective change in the interval of the affected surveillances. The probability of instrument drift or the ability to detect a failed or drifted instrument remains unchanged. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. No system required to actuate to respond to any accident is adversely affected by the proposed changes. Since each system's operability is not affected, the margin of safety associated with these systems will not be significantly reduced.

Category 2: The associated systems will be required for monitoring purposes only and provide no automatic safety actuation function. Redundant, or redundant and alternate monitoring instrumentation will still be available as required by the technical specifications during the performance of the associated surveillances. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. There will be no effective change in the intervals of the affected surveillances. The probability of instrument drift or the ability to detect a failed or drifted instrument remains unchanged. No plant transients will be initiated as a result of the proposed changes. No initiator of any accident previously evaluated is adversely affected. No system required to actuate to respond to any accident is adversely affected by the proposed changes. Therefore, the associated systems' functions in accident mitigation are not affected, and no margin of safety will be significantly reduced.

Category 3: There will be no change in the system configuration or plant operating conditions during the performance of associated surveillances, the associated system's ability to perform required safety functions will not be affected, whether the plant is shutdown for refueling or shutdown for maintenance. The surveillances performed to meet the requirements of NMP1 Technical Specifications Tables 4.6.2a Parameter 8 and 4.6.2g Parameter 6 may also be completed in concurrence with a unit shutdown. The only difference between the proposed changes and the normal unit shutdown sequence is that the mode switch may be taken to ''Shutdown'' in order to scram the plant. The response of the plant is the same as it is under the current plant shutdown procedures. There are no other differences in testing techniques or testing criteria from those previously required by the NMP1 Technical Specifications. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. There will be no effective change in the intervals of the affected surveillances. The probability of instrument drift or the ability to detect a failed or drifted instrument remains unchanged. No unexpected plant transients will be initiated as a result of the proposed changes. No initiator of any accident if adversely affected. No system required to actuate to respond to any accident previously evaluated is adversely affected by the proposed changes. Therefore, no margin of safety will be significantly reduced.

Category 4: The associated system or equipment will be isolated or out of service during the performance of the required surveillances. The associated surveillances will be performed during quarterly valve stroking, at which time the system or equipment will already be out of service. No physical change to the plant design, materials, or standards is involved. No change to instrumentation operating characteristics outside current tolerances will be made. There will be no effective change in the intervals of the affected surveillances. The probability of instrument drift or the ability to detect a failed or drifted instrument remains unchanged. No plant transients will be initiated as a result of the proposed changes. No accident initiator or failure of a different type than identified in the UFSAR is introduced. Therefore, no margin of safety will be significantly reduced.

The correction of the typographical errors is administrative only and has no affect on plant systems or procedures. In all cases, the changes will not adversely affect any accident precursor or initiator of any accident and, therefore, the changes do not introduce any new failure modes or conditions that may create a new or different accident. None of the proposed changes involve physical modification of the plant or alterations to any accident or transient analysis. Therefore, for this and the above reasons, these proposed changes do not involve any significant reduction in the margin of safety.

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

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: S. Singh Bajwa, Acting Director.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: October 16, 1996.

Description of amendment request: The proposed amendment would change certain requirements stated in Technical Specification 3/4.8.1, ''AC Sources''. The requirements are related to the emergency diesel generators (EDGs). The proposed changes would:

1. Increase the EDG fuel storage system minimum volume requirements specified in Limiting Condition for Operation 3.8.1.1.b.2;
2. Add a footnote applicable to Surveillance Requirement 4.8.1.1.2.f to qualify the words during shutdown. The footnote would allow the option of performing selected surveillances, or portions thereof, during conditions or modes other than shutdown;
3. Delete from Surveillance Requirement 4.8.1.1.2.f.14 the requirement to verify that the cooling tower fans start automatically on a Tower Actuation signal; and
4. Delete Surveillance Requirement 4.8.1.1.2.h.2 which specifies performing a periodic pressure test on the ASME Code Class 3 diesel fuel oil piping.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)).

1. Limiting Condition for Operation 3.8.1.1.b.2

The proposed change increases the minimum EDG fuel oil storage requirement to account for various factors that may affect the fuel consumption rate. The revised storage requirement reflects actual EDG test data and accounts for external variables including fuel oil specific gravity, heating value of the fuel, and ambient conditions. The proposed increase in the minimum volume storage requirement is conservative and ensures that there will be at least a 7 day supply of fuel oil stored for each EDG to meet the maximum Engineered Safety Feature load requirements following a loss of power and a design basis accident as described in Updated Final Safety Analysis Report (UFSAR) Section 9.5.4.1, Diesel Generator Fuel Oil Storage and Transfer System—Design Basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Surveillance Requirement 4.8.1.1.2.f

The proposed change qualifies the requirement to perform EDG surveillance requirements ''during shutdown''. Because the terms Hot Shutdown and Cold Shutdown are defined in the TSs as operating modes or conditions, the requirement to perform certain surveillances during shutdown may be misinterpreted, as noted in NRC Generic Letter 91–04. The proposed footnote would permit certain maintenance and testing activities to be performed during conditions or modes other than shutdown. The proposed footnote to Surveillance Requirement 4.8.1.1.2.f would not alter the intent or the method by which the surveillances are conducted, and the acceptance criteria for the surveillances would be unchanged. The footnote would not degrade the ability of the EDGs to perform their intended function, and it would not affect the response of the EDGs to a loss of power as described in the UFSAR. Since plant response to an accident would not change and since failure of an EDG could not initiate any of the accidents evaluated in the UFSAR, the proposed footnote would not alter the probability or consequences of an accident previously analyzed.

3. Surveillance Requirement 4.8.1.1.2.f.14

The cooling tower functions as the ultimate heat sink following a seismic event which results in blockage of the circulating water tunnels and therefore a loss of service water. Amendment 18 eliminated the requirement for automatic start of the cooling tower fans; therefore, the automatic-start function for the cooling tower fans has been defeated by placing the control switch in ''Pull-to-Lock''. The proposed change to delete the automatic fan start reference from Surveillance Requirement 4.8.1.1.2.f.14 is administrative only to correct an oversight since the requirement should have been deleted with the issuance of Amendment 18. The proposed deletion does not affect the manner by which the facility is operated or involve any changes to equipment or features which affect the operational characteristics of the facility. Since there is no change to the facility or operating procedures, there is no affect upon the probability or consequences of any accident previously analyzed.

4. Surveillance Requirement 4.8.1.1.2.h.2

The ASME Code, Section XI, including applicable ASME Code Cases as authorized by the NRC, provides alternate test methods to use in lieu of a 110% hydrostatic pressure test that is not practical to perform on the EDG fuel oil system as currently designed. With the proposed deletion of Surveillance Requirement 4.8.1.1.2.h.2, the provisions of Surveillance Requirement 4.0.5 and the ASME Code along with NRC-authorized Code Cases would be utilized as an equivalent testing requirement to ensure the continued integrity of the diesel fuel oil system. Therefore, since the reliability of the EDG fuel oil system will not be reduced, the probability or consequences of any accident previously evaluated is not increased.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)).

1. Limiting Condition for Operation 3.8.1.1.b.2

The proposed minimum fuel storage requirement has been developed using actual EDG performance data and accounting for possible variations in fuel oil specific gravity, heating value of the fuel, and ambient conditions. The proposed change will provide additional assurance that there will be at least a 7 day supply of fuel oil to meet the maximum Engineered Safety Feature load requirements following a loss of power and a design basis accident. The amount of fuel oil stored has no effect upon the initiation of any accident sequence, therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

2. Surveillance Requirement 4.8.1.1.2.f

The proposed change to allow the option (as supported by a 10 CFR 50.59 safety evaluation) of performing selected surveillance tests, or portions thereof, during conditions or modes other than during shutdown does not affect the operation or response of any plant equipment, including the EDGs, or introduce any new failure mechanism. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Surveillance Requirement 4.8.1.1.2.f.14

Amendment 18 to the Seabrook Station Operating License approved the change in the cooling tower operating mode from automatic actuation to manual actuation. The proposed change to Surveillance Requirement 4.8.1.1.2.f.14 does not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because it does not affect the manner by which the facility has been operated since Amendment 18 was issued, involve any changes to equipment or features which affect the operational characteristics of the facility, or introduce a new failure mode. The proposed change merely corrects an oversight in that the requirement should have been deleted when Amendment 18 was issued.

4. Surveillance Requirement 4.8.1.1.2.h.2

The change does not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because it does not affect the manner by which the facility is operated as assumed in the design analysis or Safety Evaluation, involve any changes to equipment or features which affect the operational characteristics of the facility, or introduce a new failure mode. The proposed change merely provides a practical alternate test method using methods acceptable per Section XI of the ASME Code, applicable ASME Code Cases as authorized by the NRC, and Regulatory Guide (RG) 1.137, ‘‘Fuel-Oil Systems at Nuclear Power Plants,’’ Revision 1, October 1979. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)).

1. Limiting Condition for Operation 3.8.1.1.b.2

The proposed change does not reduce the ability of the EDGs to provide sufficient power for at least 7 days to meet the maximum Engineered Safety Feature load requirements following a loss of power and a design basis accident as described in UFSAR Section 9.5.4.1.

2. Surveillance Requirement 4.8.1.1.2.f

The proposed change does not reduce the ability of the EDGs to provide sufficient power to meet the maximum Engineered Safety Feature load requirements following a loss of power and a design basis accident as described in the UFSAR. Performing certain surveillances during conditions or modes other than shutdown (as supported by a 10 CFR 50.59 safety evaluation) does not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because it does not affect the manner by which the facility is operated as assumed in the design analysis or Safety Evaluation, involve any changes to equipment or features which affect the operational characteristics of the facility. The proposed change will continue to ensure the reliability of the EDGs to perform their intended function.

3. Surveillance Requirement 4.8.1.1.2.f.14

The change does not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because it does not affect the manner by which the facility has operated since Amendment 18 was issued, involve any changes to equipment or features which affect the operational characteristics of the facility, or introduce a new failure mode. The proposed change merely corrects an oversight in that the requirement should have been deleted when Amendment 18 was issued.

4. Surveillance Requirement 4.8.1.1.2.h.2

The change does not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because it does not affect the manner by which the facility is operated or involve any changes to equipment or features which affect the operational characteristics of the facility. The proposed change will continue to ensure the reliability of the EDG fuel oil system. The proposed change merely provides a practical alternate test method using methods acceptable per Section XI of the ASME Code, applicable ASME Code Cases as authorized by the NRC, and Regulatory Guide (RG) 1.137, ‘‘Fuel-Oil Systems at Nuclear Power Plants,’’ Revision 1, October 1979.

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: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities

Service Company, Post Office Box 270, Hartford CT 06141–0270.

NRC Project Director: S. Singh Bajwa, Acting.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: October 17, 1996.

Description of amendment request: The proposed amendment would delete certain instrumentation requirements stated in Technical Specification (TS) 3/4.3, Instrumentation. The deleted requirements would be relocated to the Seabrook Station Technical Requirements Manual (SSTR). The associated Bases for the deleted TS requirements will be deleted also, but they will not be incorporated into the SSTR. The following Limiting Conditions for Operation (LCO) and associated Surveillance Requirements (SRs) would be relocated to the SSTR:

Technical specification	Title
LCO—3.3.3.2	Incore Detector System.
LCO—3.3.3.3 and associated SRs & Tables.	Seismic Instrumentation.
LCO—3.3.3.4 and associated SRs & Tables.	Meteorological Instrumentation
LCO—3.3.4 and associated SRs.	Turbine Overspeed Protection.

The proposed amendment would also delete (without relocating to the SSTR) the reference to the location of the meteorological tower from TS 5.5.

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed changes do not involve any physical changes to the plant, do not alter the way any structure, system or component functions, do not modify the manner in which the plant is operated, do not impact the physical protective boundaries of the plant, and do not decrease the effectiveness of administrative controls for assuring safe operation of the facility. The instrumentation-related systems are not considered a design feature or an operating restriction that is an initial condition of a design basis accident or transient analysis, nor do they function in any way to mitigate the consequences of a design basis accident or transient.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the proposed changes do not involve any physical changes to the plant, do not alter the way any structure, system or component functions, do not modify the manner in which the plant is operated, do not impact the physical protective boundaries of the plant, and do not decrease the effectiveness of administrative controls for assuring safe operation of the facility.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not involve any physical changes to the plant, do not alter the way any structure, system or component functions, do not modify the manner in which the plant is operated, do not impact the physical protective boundaries of the plant, and do not decrease the effectiveness of administrative controls for assuring safe operation of the facility. Further, the proposed changes do not affect the ability of systems, structures or components important to safety to perform their intended function.

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: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141–0270.

NRC Project Director: S. Singh Bajwa, Acting.

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

Date of amendment request: March 20, 1996 and as supplemented on July 25, 1996.

Description of amendment request: The amendments would modify the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications to change the ''open'' logic for the high pressure core injection (HPCI) suction valves HV–155/255–F042 in order to eliminate the HPCI pump auto-transfer on high suppression pool level.

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.

Based on the following discussion for the containment, reactor building, HPCI and RCIC [reactor core isolation cooling] systems, and the safety-related valves in piping connected to the suppression pool, the proposed action does not increase the probability or consequences of an accident previously evaluated. Primary Containment and Reactor Building Safety-Related Systems, Structures, and Components Affected by LOCA/SRV [Loss-of-coolant-accident/safety relief valve] Hydrodynamic Loads

As discussed in the Safety Assessment for this change, elimination of the HPCI auto suction transfer on high suppression pool level will allow higher suppression pool water levels in accidents and transients which involve HPCI operation. The impact of the higher suppression pool levels were examined for the following design-basis accidents and transients:

Loss of Coolant Accidents inside containment (FSAR [Final Safety Analysis Report] *6.2.1.1.3.3),

Inadvertent Safety/Relief valve opening (FSAR *15.1.4),

Primary system break outside containment (FSAR *3.6A),

Inadvertent HPCI initiation (FSAR *15.5.1),

Loss of feedwater flow (FSAR *15.2.7),

Loss of Offsite AC Power (FSAR *15.2.6),

Loss of Main Condenser vacuum (FSAR *15.2.5),

Inadvertent MSIV closure (FSAR *15.2.4),

Turbine trip (with and without bypass) (FSAR *15.2.3),

Generator Load Rejection (with and without bypass), (FSAR *15.2.2), and

Pressure regulator failure-closed/open (FSAR *15.2.1 & 15.1.3).

These accidents and transients were selected for evaluation because they involve an initiation of the HPCI system either inadvertently or as a result of a decrease in vessel inventory and/or coolant level. Two special events, ATWS and SBO, are also considered along with the design basis events listed above.

It was concluded that design-basis SRV and LOCA loads envelop the loads expected with the proposed change. Therefore, the proposed change does not increase the failure probability of any primary containment or reactor building structure, system or component which is affected by LOCA/SRV hydrodynamic loads. The major findings which lead to this conclusion about SRV and LOCA loads are summarized below:

DBA [design basis accident] dynamic pressure loads are based on a maximum initial suppression pool level of 24 feet. The proposed modification to the HPCI suction transfer logic does not affect the initial pool level or the initial suppression chamber air space volume. During normal plant operation, suppression pool level (and hence suppression chamber air space volume) is controlled by Technical Specification requirements.

For LOCAs other than the DBA, the containment is designed for ADS [automatic depressurization system] blowdown loads in combination with the LOCA loads. For an intermediate break, the proposed HPCI modification does allow suppression pool level to exceed 24 feet by a small amount. ADS loads are, however, independent of suppression pool level when the downcomer vents are cleared. Therefore, the proposed modification has no influence on ADS hydrodynamic loads for an intermediate break.

For small breaks, HPCI injection prevents ADS actuation. Nevertheless, SRV actuations occur during the RPV [reactor pressure vessel] cooldown. Downcomer vents are opened in the beginning part of the accident, but close later on as the break enthalpy decreases. When the downcomer vents are cleared, the level inside the SRV tailpipe is not influenced by pool level, and therefore, the SRV hydrodynamic loads are unaffected by the proposed modification. During the phase of the accident in which the downcomer vents are sealed with water, there are no wetwell LOCA hydrodynamic loads, but the SRV loads are dependent on SP [suppression pool] water level. In this case, SRV loads are acceptable because SP water level is always below the Load Limit curve.

ADS actuation would be required in the event of a HPCI failure during a small-break accident. If HPCI fails during the phase of the accident in which the downcomer vents are cleared, then ADS loads would be acceptable because water level (and air volume) within the SRV tailpipes is independent of pool level. Even if HPCI failure occurs in the latter part of the accident where the downcomer vents are sealed, ADS loads are acceptable because water level is always well below the Load Limit curve.

Under non-LOCA conditions, the containment is designed for simultaneous actuation of all 16 SRVs. The Load Limit Line defines the acceptable operating region, in terms of reactor pressure and suppression pool level, for actuation of all 16 SRVs. Following a plant transient involving HPCI operation, the suppression pool level is always below the Load Limit curve, and only a small number of SRVs actuate to remove decay heat from the reactor.

HPCI System

The proposed change does not increase the probability of an equipment malfunction in the HPCI system. In fact, the change eliminates the potential failure of the HPCI suction auto-transfer on high suppression pool level since that logic is removed. Potential spurious auto-transfer associated with high suppression pool logic is also eliminated. HPCI suction auto-transfer on low CST [condensate storage tank] level and its potential to fail are unchanged by this change. Also, the change does not affect the manual suction transfer from the CST to the suppression pool.

As discussed in the safety assessment for this change, the proposed change has no adverse effects on HPCI valves, pump, or turbine. Therefore, elimination of the HPCI suction auto transfer logic (on high suppression pool level) does not increase the probability of a HPCI malfunction. The consequence of a HPCI failure in a design-basis accident is evaluated in NEDC–32071P Rev.1, ''Susquehanna Steam Electric Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis.'' With regard to the fuel, the consequence of a HPCI failure is unaffected by the proposed change.

If HPCI fails in a design-basis small break accident, ADS actuation would be required. ADS loads continue to be enveloped by design loads with the proposed change. Therefore, the proposed change does not increase the consequences of a HPCI failure.

HPCI Relay Panel 1C620(2C620) & 250 V DC Control Center 1D264(2D264)

On a component level, the failure probability and consequences of failure associated with the AX [auxiliary] relay in 250 VDC Control Center 1D264 (2D264) are eliminated because the relay is disconnected and removed by this modification. Since the control functions of K19 in panel 1C620 (2C620) have been eliminated, the failure of the relay has no effect on HPCI suction valve F042 operation.

The 250 VDC Control Center 1D264 (2D264) and HPCI Relay Panel 1C620 (2C620) both receive power from battery systems during Station Blackout. Removal of the relay from 250 VDC Control Center 1D264 (2D264) and the replacement of the relay in HPCI Relay Panel 1C620 (2C620) decreases the load on the battery systems by a small amount. The change in battery load and line voltage drop is negligible and is documented in applicable calculations. Dynamic qualification of the subject equipment is not adversely affected by this modification as documented in applicable calculations.

RCIC Turbine

As discussed in the safety assessment for this change, RCIC is used to provide coolant makeup following a reactor vessel isolation and for an Appendix R shutdown scenario. The Appendix R event also assumes the reactor vessel is isolated. These events are discussed in Section 15.2.4 of the FSAR and in the FPRR [fire protection review report]. The proposed change has no adverse effects on RCIC turbine operation following a MSIV [main steam isolation valve] closure (see discussion in the safety assessment for this change [letter dated March 20, 1996, as supplemented July 25, 1996]). Therefore, there is no increase in the RCIC failure probability for the MSIV-closure event or the Appendix R shutdown scenario. The consequence of RCIC failure is unchanged by the proposed modification; if RCIC fails, HPCI is available as a backup system.[1] [All footnotes are listed at the end of the no signficant hazards basis section.]

Although RCIC is not designed for mitigation of a small break accident, the effect of the proposed change on RCIC turbine operation for such an accident was evaluated in the safety assessment for this change. The assessment concludes that the proposed change has no adverse effects on RCIC operation, and therefore, there is no increase in RCIC failure probability during a small break accident. Failure of RCIC in a small break accident would require ADS initiation only for a particular break flow which is slightly greater than HPCI injection capability. But ADS initiation has already been considered when evaluating the consequences of HPCI failure during a small break accident.

Safety-Related Valves on Piping Connected to Suppression Chamber

MOVs [motor operated valves]—The proposed change could potentially lead to a maximum suppression pool level of 26 feet in a design-basis accident. This is 2 feet above the maximum design level of 24 feet. As discussed in the safety assessment for this change, this is equivalent to a pressure increase of 0.86 psi at the bottom of the suppression pool. This small pressure increase has negligible effect on valve operation, and therefore, there is no increase in the probability of a failure or malfunction of valves in piping connected to the suppression pool.

Vacuum Breakers—Allowing suppression pool level to potentially increase to 26 feet in a design-basis accident does not affect the failure probability of downcomer-vent vacuum breakers because the level is well below the vacuum breaker elevation of 42 feet.

SRVs/Tailpipes—As discussed in the safety assessment for this change, the increased suppression pool level associated with the proposed change does not have any adverse effect on SRV operation or on the structural integrity of the SRV tailpipe.

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

Based on the following discussion for the containment, reactor building, HPCI and RCIC systems, and the safety-related valves in piping connected to the suppression pool, the proposed action does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The following discussion concerning the impact of the change on the primary containment, the reactor building, the HPCI system, and safety-related valves, provides the basis for this conclusion.

Primary Containment and Reactor Building Safety-Related Systems, Structures, and Components Affected by LOCA/SRV Hydrodynamic Loads

The HPCI suction transfer logic is not necessary to maintain LOCA loads within design limits because these dynamic pressure loads are characterized in terms of the SP level at the initiation of the accident. That is, LOCA blowdown tests were conducted without the removal of water from the suppression chamber section of the test tank.[2] The increase in pool level realized during these tests was proto-typical of the pool level increase expected at Susquehanna. Removal of the HPCI suction transfer logic on high pool level does not affect suppression pool level at the initiation of a DBA.[3]

In addition, the HPCI suction transfer logic is not necessary to maintain SRV/ADS blowdown loads within design limits. SRV dynamic pressure loads consist of two components: air clearing loads and steam condensation loads. The steam condensation loads are bounded by the more severe air clearing loads which are caused by gas bubble oscillations following the expulsion of noncondensible gas from the SRV tailpipe. Air clearing loads are a function of reactor pressure and water level inside the SRV tailpipe.

Depending on the break size and location, the downcomer vents may be cleared for the entire time that HPCI is operating, or they may reseal in the latter part of the accident. When the downcomer vents are cleared, the level inside the SRV tailpipe is depressed to the elevation coinciding with the bottom of the downcomer pipes, and it is therefore decoupled from the rising suppression pool level. In this situation SRV air-clearing loads are unaffected by the proposed change.

When the downcomer vents are sealed with water, the Load Limit line can be used to determine if SRV/ADS loads are enveloped by design loads. For the most limiting event, which is the small break LOCA, the overall safety margin increases as pool level rises during the event. This is because the decrease in reactor pressure more than offsets the adverse effects associated with the rise in pool level.

Since LOCA and SRV dynamic loads remain bounded by design loads, dynamic loading of primary containment and reactor building structures, systems, and components are unaffected by the proposed change. Therefore, with respect to dynamic loads, the proposed change does not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR [Safety Analysis Report].

HPCI System

There are no new HPCI turbine failure modes introduced by the higher suppression pool levels which can occur with the proposed change. Turbine exhaust pressure remains well below the design limit of 65 psia. In addition, the higher pool level does not create the possibility of water hammer damage to the turbine discharge piping. If the operator fails to control RPV level less than +54″ (single operator error) in the long-term part of the small-break accident when suppression pool level is greater than 25.6 feet, leakage through check valve F049 is such that it will be contained well within the volume of the turbine-discharge-line drain pot. Note that suppression pool level is limited to 26 feet by operator action. Furthermore, suppression pool level can reach 26 feet only for a particular range of small breaks, and for this range of small breaks, suppression pool level would exceed 25.6 feet for only approximately 10 minutes of the accident duration. This corresponds to about 10% of the time that HPCI is operating. Thus it is very unlikely that HPCI would trip with pool level greater than 25.6 feet.

If check valve F049 is failed during the small-break accident (single equipment failure), the turbine exhaust line would become flooded if the HPCI system tripped during the 10 minute interval when suppression pool level greater than 26 feet; however, it is not necessary to postulate an operator error (failure to control RPV level less than +54″) along with the check valve failure. A small break accident with failure of check valve F049 and failure of the operator to control RPV level as required by the EOPs [emergency operating procedures], in a narrow time interval during the long-term part of the accident, is beyond the plant design basis.

A new type of malfunction does not occur even in the beyond-design-basis condition where failure of check valve F049 is considered along with failure of the operator to control RPV level less than 54″ in the narrow time interval when pool level is greater than 25.6. With these failures, the turbine exhaust piping will become flooded, and the system may fail on restart. The General Electric Company has performed an analysis to determine the consequences of a HPCI start with flooding of the turbine and adjacent exhaust line.[4] The analysis, which addresses a potential design deficiency in the HPCI barometric condenser, shows that the containment penetration head fitting and interface piping will not fail as a result of the water hammer associated with the HPCI start. Since failure of the HPCI system is already considered in the plant design-basis accident analysis; this is not a different type of malfunction than that already considered.

HPCI Relay Panel 1C620(2C620) & 250 V DC Control Center 1D264(2D264)

No new failure modes are introduced by the hardware changes in the 250 VDC Control Center 1D264 (2D264) and HPCI Relay Panel 1C620 (2C620). Some failure modes are eliminated by the proposed change. Specifically, the potential failure of the HPCI suction auto-transfer on high suppression pool level is eliminated since that logic is removed. Potential spurious auto-transfer associated with high suppression pool logic is also eliminated. HPCI suction auto-transfer on low CST level and its potential to fail are unchanged by this change.

On a component level, potential failure modes for the AX relay in 250 VDC Control Center 1D264 (2D264) are eliminated by this modification because the relay is disconnected and removed by this change. The potential failure modes for the relay K19 in panel 1C620 (2C620) are unchanged. Since the control functions of K19 have been eliminated, the failure of the relay has no effect on HPCI suction valve F042 operation.

Removal of the relay from 250 VDC Control Center 1D264 (2D264) and the replacement of the relay in the HPCI Relay Panel 1C620 (2C620) changes the load on the battery systems by a small amount. The change in battery load and change in line voltage drop are negligible and they do not adversely affect the performance of the panels or battery systems. In addition, seismic qualification of the panels is not adversely affected by this change.

RCIC Turbine

As discussed in the safety assessment for this change, the proposed change has no adverse effects on RCIC turbine operation. Therefore, the proposed change cannot result in a new RCIC failure mode.

Safety-Related Valves on Piping Connected to Suppression Chamber

MOVs—The increased suppression pool water level which can occur as a result of the proposed change does not create a failure mechanism for safety-related valves on piping connected to the suppression pool. The pressure differential for any valve on piping connected to the suppression pool will increase by at most 0.86 psi. This change in differential pressure has negligible effect on valve operation.

Vacuum Breakers—The proposed change cannot lead to malfunction of the downcomer-vent vacuum breakers as the maximum level expected in a design-basis event is 26 feet, and the vacuum breakers are located at 42 feet above the suppression pool floor.

SRVs/Tailpipes—There is no interaction between increased suppression pool level and SRV operation since the flow through the SRVs is choked and therefore decoupled from downstream conditions. Also, the increased suppression pool level cannot lead to failure of the SRV tailpipe because the potential level increase is well below the SRV Tailpipe Level Limit.[5] If suppression pool water level is below this limit, there is no concern of tailpipe failure due to overpressurization. The minimum value of the SRV Tailpipe Level Limit is 35 feet.[6] This is 9 feet above the maximum level expected in a design-basis accident. For beyond-design-basis events, SRV tailpipe integrity is protected by the EOP requirement to depressurize the reactor on the SRV Tailpipe Level Limit.

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

Based on the following discussion for the containment, reactor building, HPCI and RCIC system, and the safety-related valves in piping connected to the suppression pool, the proposed action does not involve a significant reduction in a margin of safety.

HPCI System

The HPCI Technical Specifications ensure that the system is capable of providing adequate core cooling to limit clad temperatures in the event of a small break LOCA which does not result in rapid depressurization of the RPV (Technical Specification Section 3/4.5.1 & 3/4.5.2). The proposed change has no adverse affects on the injection capability of the HPCI system. Therefore, the safety function of the system is not degraded, and there is no reduction in the margin of safety as defined in the basis for the HPCI Technical Specifications.

Primary Containment and Reactor Building Safety-Related Systems, Structures, and Components Affected by LOCA/SRV Hydrodynamic Loads

Removal of the HPCI auto suction transfer on high suppression pool level does not affect the Technical Specification requirement to maintain suppression pool water level between 22 and 24 feet (Technical Specification 3.6.2.1). Therefore, the maximum containment pressure during the design-basis accident is unaffected by the proposed change, and there can be no reduction in the margin of safety as defined in the basis for Technical Specification

3.6.2.1. Furthermore, a detailed examination of the reactor and containment response under accident and transient conditions involving HPCI operation found no situations where the auto suction transfer was necessary to maintain LOCA and SRV loads within the design basis envelope. Therefore, from the standpoint of LOCA/SRV hydrodynamic loads, the proposed change does not reduce the margin of safety for any primary containment or reactor building structure, system, or component.

RCIC Turbine

The basis for Technical Specification 3.7.3 states that the RCIC system is provided to assure adequate core cooling in the event of a reactor isolation with loss of feedwater flow. The proposed change does not prohibit RCIC from performing this function, nor does it degrade in any way the core cooling capability of RCIC. Therefore, there is no reduction in the margin of safety as defined in the basis for Technical Specification 3.7.3.

Safety-Related Valves on Piping Connected to Suppression Pool

MOVs—The increase in suppression pool water level which can occur as a result of the proposed change does not reduce the margin of safety for safety-related valves on piping connected to the suppression pool. The pressure differential for any valve on piping connected to the suppression pool will increase by at most 0.86 psi. This change in differential pressure has negligible effect on valve operation.

Vacuum Breakers—The proposed change cannot reduce the margin of safety as discussed in the basis for Technical Specification 3.6.4 because the maximum level expected in a design-basis event is 26 feet which is well below the downcomer-vent vacuum breaker elevation of 42 feet.

SRVs/Tailpipes—There is no interaction between increased suppression pool level and SRV operation since the flow through the SRVs is choked and therefore decoupled from downstream conditions. Consequently, there is no reduction in the margin of safety as defined in the bases for Technical Specifications 3.4.2 (safety valve function) and 3.5.1.d (ADS function). Also, the increased suppression pool level does not lead to a reduction in the margin of safety for the SRV tailpipes because the tailpipes can operate safely with pool levels up to 35 feet. This is nine feet above the maximum suppression pool level that can occur in a design-basis accident with the proposed change. For beyond-design-basis events, SRV tailpipe integrity is protected by the EOP requirement to depressurize the reactor on the SRV Tailpipe Level Limit.[7]

HPCI Relay Panel 1C620(2C620) & 250 V DC Control Center 1D264(2D264)

As discussed previously, removal of the relay from 250 VDC Control Center 1D264 (2D264) and the replacement of the relay in the HPCI Relay Panel 1C620 (2C620) changes the load on the battery systems by a small amount. The change in battery load and change in line voltage drop are negligible and therefore they do not reduce the margin of safety for the panels or battery systems. In addition, seismic qualification of the panels is not adversely affected by this change so there is no reduction in the margin of safety for seismic events.

1. DBD041, Rev. 0, p. 1. [design basis document for RCIC system]
2. SSES DAR [design assessment report for suppression pool hydrodynamic loads], Section 9.4.1
3. Suppression pool level must be maintained less than 24 feet in accordance with Technical Specification 3.6.2.1.a.
4. GKR–03–001, ''NRC and Utility Notification of Closeout of GE PRC92–05, Potential Design Deficiency on HPCI,'' January 6, 1993 [GE letter to PP&L regarding closure of HPCI design issue].
5. This limit is defined in EO–100/200–103 [emergency operating procedure]
6. Bechtel Calculations PUP–15598–S2 & PUP–15598–S6, and PLE–15315 (March 2, 1992)
7. The limit is defined in EO–100/200–103 [emergency operating procedure]

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: October 7, 1996.

Description of amendment request: The amendments would modify the Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications by revising the trip setpoints and allowable values for the secondary containment isolation ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor in Table 3.3.2–2. The change would enhance the operational efficiency of plant operations by eliminating compensatory measures which prevent spurious secondary containment isolations, and initiation of the standby gas treatment system (SGTS) and recirculation system during refueling activities. This change would also allow for the use of the hydrogen water chemistry system during operation.

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

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

The proposed change to the trip setpoints and allowable values to the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor does not involve a significant increase in the probability or consequences of an accident previously evaluated. The design basis for the monitors is to monitor radiation in the unfiltered air from the Zone III exhaust system to provide signals which isolate the Zone III of the secondary containment on a high radiation condition, and to initiate SGTS and the Recirculation system to limit offsite doses to maintain regulatory requirements.

The original setpoints for these monitors were based upon normal radiological operating conditions and were set at a value to preclude spurious design actuations by these monitors during normal plant operations. However, the monitors are designed to detect radiation associated with certain postulated accident conditions. As required by the Technical specifications the monitors are operable when conditions exist that may result in fuel damage events, and therefore, will perform their design basis function. Consequently, an increase to the trip setpoints and allowable values is warranted since the existing setpoints, which are conservatively based on normal radiological operating conditions, are not related to the design basis of the monitors. Therefore, based upon the design basis of the monitors, an increase to the trip setpoints and allowable values will not result in a decrease of the safety function of the monitors but will make the trip setpoints and allowable values consistent with the design basis.

Based on the design basis of these monitors, revised analytical limits were derived reflecting the accident function of the monitors. The analytical limit calculations utilized FSAR realistic source terms, instead of the worst case source terms utilized for 10CFR [Part] 100 compliance. Use of the realistic source terms results in conservative analytical limits.

The ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' [monitor] are required to be OPERABLE during CORE ALTERATIONS (except for single control rod movements unless performing TS 3.10.3), operations with the potential for draining the reactor vessel, and handling of irradiated fuel in the secondary containment. The ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor is required to be operable during handling of irradiated fuel. These Technical Specification applicable operational conditions for the monitors are not affected since this proposed revision only revises the trip setpoints and allowable values to be consistent with the design bases of the monitors.

For the reasons stated above the revisions to the trip setpoints and allowable values to the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor in Technical Specification.

Table 3.3.2–2 can be implemented without a significant increase in the probability or consequence of an accident previously evaluated.

II. This proposal does not create the possibility of a new or different kind of accident previously evaluated.

The proposed change to the trip setpoints and allowable values for the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The monitors are designed to limit the release of airborne radioactivity in the secondary containment Zone III exhaust system by isolating Zone III, initiating [the] SGTS and initiating the Recirculation System on high radiation resulting from fuel handling accidents. Therefore, the design basis for these monitors is to monitor radiation in the unfiltered air from the Zone III exhaust system, and provide signals to limit offsite doses to maintain regulatory requirements. Zone III includes the Refueling Floor and can include the Railroad Access Shaft during certain alignments. These radiation monitors are not provided for occupational protection associated with operational radiation doses. The proposed revision does not affect the design basis of the monitors nor the kind of accident associated with the basis; therefore, no potential to create a new or different accident exists.

For the reasons stated above the revisions to the trip setpoints and allowable values to the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor in Technical Specification Table 3.3.2–2 can be implemented without creating the possibility of a new or different kind of accident from any accident previously evaluated.

III. This proposal does not involve a significant reduction on a margin of safety.

The proposed change to the trip setpoints and allowable values for the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor does not involve a significant reduction in a margin of safety.

The monitors are designed to limit the release of airborne radioactivity in the secondary containment Zone III exhaust system by isolating Zone III, initiating [the] SGTS and initiating the Recirculation System on high radiation resulting from fuel handling accidents. Therefore, the design basis for these monitors is to monitor radiation in the unfiltered air from the Zone III exhaust system, and provide signals to limit offsite doses to maintain regulatory requirements. Zone III includes the Refueling Floor and can include the Railroad Access Shaft during certain alignments. These radiation monitors are not provided for occupational protection associated with operational radiation doses. However, the original setpoints for these monitors were conservatively based upon normal radiological operating conditions and were set at a value to preclude spurious design actuation by these monitors during normal plant operations. The calculations performed to support the trip setpoint and allowable value revisions concluded that the change will maintain offsite doses within the 10CFR100 limits. The ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' are required to be OPERABLE during CORE ALTERATIONS (except for single control rod movements unless performing TS 3.10.3), operations with the potential for draining the reactor vessel, and handling of irradiated fuel in the secondary containment. The ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor is required to be operable during handling of irradiated fuel. These Technical Specification applicable operational conditions for the monitors are not affected since the proposed revision only revises the trip setpoints and allowable values to be consistent with the design bases of the monitors.

The proposed revisions to the trip setpoints and allowable values, in addition to being based on the appropriate accident conditions, were also developed utilizing standard setpoint change methodologies that consider instrument and calibration accuracies and instrument drift tolerances. This provides added conservatism to assure that the revised trip setpoints and allowable values are not exceeded.

For the reasons stated above the revisions to the trip setpoints and allowable values to the ''Refuel Floor High Exhaust Duct Radiation—High'' monitor, the ''Railroad Access Shaft Exhaust Duct Radiation—High'' monitor, and the ''Refuel Floor Wall Exhaust Duct Radiation—High'' monitor in Technical Specification Table 3.3.2–2 can be implemented without involving a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: November 25, 1996.

Description of amendment request: The proposed Technical Specifications (TS) changes would revise the wording in TS Section 4.8.1.1.2.e.2 and the associated TS Bases Section 3/4.8 to remove the specific reference to the Residual Heat Removal pump motor and its corresponding kW rating value, and replace it with wording consistent with that specified in the Improved TS (i.e., NUREG–1433, Revision 1, ''Standard Technical Specifications General Electric Plants,'' dated April 1995).

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

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

The proposed TS changes do not make any physical alterations or modifications to the plant systems or equipment. The proposed changes do not adversely impact the operation of any plant equipment. The EDGs will continue to function as designed to ensure that the necessary electrical power is provided to essential plant equipment to mitigate the consequences of an accident, e.g., Loss-of-Offsite-Power (LOOP) and Loss-of-Coolant Accident LOCA) coincident with a LOOP (LOCA/LOOP). The proposed TS changes do not impact the performance testing requirements associated with the EDGs. The accident mitigating capabilities of the diesel generators and emergency loads will remain the same.

The proposed TS changes are consistent with the guidance stipulated in NUREG–1433, Revision [1], ''Standard Technical Specification General Electric Plants,'' regarding single load rejection testing of the EDGs. Specifically, the proposed changes involve revising the wording in TS Surveillance Requirement (SR) 4.8.1.1.2.e.2 to remove the specific reference to the Residual Heat Removal (RHR) pump motor and associated kW loading value (992 kW), and replace it with wording indicating that the EDGs must be capable of rejecting the single largest post-accident load, which is consistent with NUREG–1433, Revision 1, guidance. The proposed changes will also provide additional flexibility for future plant maintenance activities.

Each EDG will continue to be tested by rejecting a load of greater than or equal to that of its single largest post-accident load while maintaining voltage and frequency within the current specified parameters. The RHR pump motors are currently used in performing the EDG single load rejection testing. The RHR pump motors will continued [sic] [continue] to be used in performing the surveillance testing since they are the single largest post-accident electrical load. The consequences of a malfunction of equipment are not affected. Failure of a EDG or its safety-related loads is bounded by the loss of a Class 1E electrical power division which has been previously evaluated as discussed in LGS Updated Final Safety Analysis Report (UFSAR) Sections 8.1.5.2.e and 8.3.1.1.3.

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

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

The proposed TS changes do not make any physical alterations or modifications to the plant systems or equipment. The proposed changes do not adversely impact the operation of any plant equipment. The EDGs will continue to function as designed to provide essential electrical power to mitigate the consequences of an accident. The proposed TS changes are consistent with the guidance stipulated in NUREG–1433, Revision 1, regarding single load rejection testing of the EDGs. The proposed changes do not introduce any new accidents or transients. The proposed TS changes will provide additional flexibility for future maintenance activities. The proposed changes do not alter any EDG testing requirements or frequencies. The RHR pump motors are currently used in performing the EDG single load rejection testing. The RHR pump motors will continue to be used in performing the surveillance testing since they are the single largest post-accident electrical load. The operation of the EDGs and their corresponding safety-related electrical loads remain unchanged as a result of the proposed TS changes.

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

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

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed TS changes are consistent with the guidance stipulated in NUREG–1433, Revision 1, ‘‘Standard Technical Specification General Electric Plants,’’ regarding single load rejection testing of the EDGs. The proposed TS changes will provide additional flexibility for future plant maintenance activities. The EDGs will continue to function as designed to provide essential electrical power to mitigate the consequences of an accident. The operation of the EDGs and their corresponding safety-related electrical loads remain unchanged as a result of the proposed TS changes.

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

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendments request: November 15, 1996.

Description of amendments request: The amendments would eliminate the containment systems Technical Specification 3.6.2.2. ‘‘Spray Additive System.’’ The specification would be replaced with a new emergency core cooling system Technical Specification 3.5.6 ‘‘ECCS Recirculation Fluid pH Control System.’’

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change involves replacement of concentrated NaOH injected via the containment spray system with trisodium phosphate (TSP) stored in the containment and dissolved in the sump recirculation solution to maintain acceptable post accident spray/recirculation solution chemistry. Deletion of the concentrated NaOH will eliminate a personnel hazard. The pH control system functions in response to an accident and does not involve or have any effect on any initiating event for any accident previously evaluated. Operation under the proposed amendments will continue to ensure that iodine potentially released post-LOCA [loss-of-coolant accident] is retained in the sump solution, and resultant offsite and control room thyroid doses are within the limits of 10 CFR [Part] 100 and 10 CFR [Part] 50, Appendix A, General Design Criterion [GDC] 19, respectively.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The deleted equipment is isolated from the remaining equipment by cut-and-capped piping, determinated and/or spared cables; and interfaces are analyzed to ensure the remaining required equipment meets applicable original design requirements. The new equipment (TSP and baskets) is a passive pH control system and is supported and analyzed to ensure there are no adverse interfaces (e.g., pipe break, jet impingement, seismic) with existing equipment, system, or structures.

3. The proposed change does not involve a significant reduction in a margin of safety. The slight change in recirculation solution pH maintains adequate protection against chloride and caustic induced stress corrosion cracking on mechanical systems and components, and maintains the capability of the solution to retain iodine. It does not result in a change to the hydrogen generation analysis for containment. The increased mass inside containment will have no significant impact on post-accident flood levels, recirculation solution boron concentration, or peak clad temperatures. No other operating parameters for systems, structures, or components assumed to operate in the safety analysis are changed. The offsite and control room doses meet the limits of 10 CFR [Part] 100 and GDC 19, respectively. Because the trisodium phosphate is nonvolatile and the baskets are protected with solid covers and are located slightly above the floor in the containment where access is strictly controlled, a surveillance interval of once per refueling outage provides assurance that the TSP will be available.

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

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

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

NRC Project Director: Herbert N. Berkow.

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

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

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

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

Date of amendment request: June 3, 1996, as supplemented October 23, 1996.

Description of amendment request: The proposed amendment would clarify a restriction on shutdown margin monitor operability while changing modes so that it only limits reactivity changes caused by boron dilution and rod withdrawal.

Date of publication of individual notice in Federal Register: June 20, 1996 (61 FR 31559).

Expiration date of individual notice: July 22, 1996.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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.

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

Date of application for amendments: October 4, 1996 and supplemented on November 6, 1996.

Brief description of amendments: The amendments add a Mode of Applicability to Technical Specification 3.2.3.D, Inoperable Rod Position Indicator Channels.

Date of issuance: November 25, 1996.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 176 and 163.

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

Date of initial notice in Federal Register: October 17, 1996 (61 FR 54240).

The November 6, 1996, submittal provided additional clarifying information that did not affect the Commission's initial proposed finding of no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 21, 1996.

Brief description of amendments: The amendments revise Technical Specification (TS) Section 3/4.9.6, "Manipulator Crane," to make the wording consistent with the TS Bases description and consistent with the design of the load handling equipment.

Date of issuance: November 25, 1996.

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

Amendment Nos.: 156 and 148.

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

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55031) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application of amendments: September 17, 1996 (TSC 96–01) as supplemented October 23, 1996.

Brief description of amendments: The amendments lower the maximum allowable reactor building pressure, lower the actuation setpoint for actuation of the reactor building spray system, and modify the associated TS Bases requirements.

Date of Issuance: November 25, 1996.

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

Amendment Nos.: 219, 219, 216.

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

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55031). The October 23, 1996, letter provided clarifying information that did not change the scope of the September 17, 1996, application and the initial proposed no signficant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 31, 1996, as supplemented by letters of September 5, October 22, and November 15, 20, and 21, 1996, which supersede the application submitted in the letter of May 9, 1996.

Brief description of amendment: The amendment (1) increased the safety limit minimum critical power ratio (MCPR) for two loop operation and single loop operation to 1.12 and 1.14, respectively, and (2) added two General Electric topical reports to the list of documents describing the analytical methods used to determine the core operating limits. The changes are to Section 2.1.1, Reactor Core Safety Limits, and Section 5.6.5, Core Operating Limits Report (COLR), respectively, of the Technical Specifications. This amendment would go into effect in Operating Cycle 9, at the end of the current Refueling Outage 8, and the plant will have a mixed core of Siemens Power Corporation (SPS) $9 \times 9 - 5$ and General Electric (GE) GE11 reload fuel. The licensee also changed the Bases of the Technical Specifications associated with the above amendment.

Date of issuance: November 21, 1996.

Effective date: November 21, 1996.

Amendment No: 131.

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

Date of initial notice in Federal Register: September 25, 1996.The October 22, and November 15, 20, and 21, 1996, submittals provide clarifying information that did not change the initial determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 15, 1996 (TSCR No. 244).

Brief description of amendment: The amendment revises Specification 5.3.1.B to allow the shield plug and the associated lifting hardware to be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

Date of Issuance: November 7, 1996.

Effective date: November 7, 1996, to be implemented within 30 days of issuance.

Amendment No.: 187.

Facility Operating License No. DPR–16: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20849). The Commission's related evaluation of this amendment and final determination of no significant hazards consideration addressing comments received on the proposed no significant hazards consideration determination are contained in a Safety Evaluation dated November 7, 1996.

No significant hazards consideration comments received: Yes.

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

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

Date of application for amendment: February 22, 1996, and as supplemented by letters dated July 24, October 4, November 19 and November 25, 1996.

Brief description of amendment: The amendment changes Clinton Power Station Technical Specification (TS) 3.3.8.1, ''Loss of Power Instrumentation,'' and TS 3.8.1, ''AC Sources-Operating,'' by revising the setpoint for the degraded voltage protection instrumentation and modifying or deleting other Loss of Power Instrumentation TS requirements. In addition, changes were also made to the minimum required diesel generator voltage specified for certain diesel generator surveillances.

Date of issuance: December 4, 1996.

Effective date: December 4, 1996.

Amendment No.: 110.

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18168).The letters of July 24, October 4, November 19 and November 25, 1996, provided clarifying information and did not represent significant changes from the original Federal Register notice.

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

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: October 11, 1996.

Brief description of amendments: These amendments revise Technical Specification (TS) 3.9.6, ''Refueling Water Level,'' for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The proposed change is required to restore certain provisions of the SONGS Units 2 and 3 operating practice that were not incorporated during the conversion to the improved TS (Amendment Nos. 127 and 116, dated February 9, 1996).

Date of issuance: December 3, 1996.

Effective date: December 3, 1996, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—134; Unit 3—123.

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

Date of initial notice in Federal Register: October 31, 1996 (61 FR 56251) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 3, 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.

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

Date of amendment request: March 24, 1995, as supplemented by letter dated July 26, 1996.

Brief description of amendment: The amendment revised Technical Specification (TS) Surveillance Requirement 4.5.1.1.a.1 to base accumulator operability on actual parameters (i.e., borated water volume and nitrogen cover-pressure in the tanks) vs. the absence of alarms.

Date of issuance: November 22, 1996.

Effective date: November 22, 1996, to be implemented within 30 days of issuance.

Amendment No.: 103.

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

Date of initial notice in Federal Register: April 12, 1995 (60 FR 18632) The July 26, 1996, letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1996.

No significant hazards consideration comments received: No.

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

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

For the Nuclear Regulatory Commission.

Steven A. Varga,

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

[FR Doc. 96–31944 Filed 12–17–96; 8:45 am]

BILLING CODE 7590–01–P

POSTAL RATE COMMISSION

[Docket No. A97–6; Order No. 1144]

Plevna, MO 63464: (William Ahern, et al., Petitioners); Notice and Order Accepting Appeal and Establishing Procedural Schedule Under 39 U.S.C. § 404(b)(5)

Issued December 13, 1996.

Docket Number: A97–6.

Name of Affected Post Office: Plevna, Missouri 63464.

Name(s) of Petitioner(s): William Ahern, *et al.*

Type of Determination: Closing.

Date of Filing of Appeal Papers: December 10, 1996.

Categories of Issues Apparently Raised:

1. Effect on the community [39 U.S.C. § 404(b)(2)(A)].
2. Effect on postal services [39 U.S.C. § 404(b)(2)(C)].

After the Postal Service files the administrative record and the Commission reviews it, the Commission may find that there are more legal issues than those set forth above. Or, the Commission may find that the Postal Service's determination disposes of one or more of those issues.

The Postal Reorganization Act requires that the Commission issue its decision within 120 days from the date this appeal was filed (39 U.S.C. § 404 (b)(5)). In the interest of expedition, in light of the 120-day decision schedule, the Commission may request the Postal Service to submit memoranda of law on any appropriate issue. If requested, such memoranda will be due 20 days from the issuance of the request and the Postal Service shall serve a copy of its memoranda on the petitioners. The Postal Service may incorporate by reference in its briefs or motions, any arguments presented in memoranda it previously filed in this docket. If necessary, the Commission also may ask petitioners or the Postal Service for more information.

The Commission Orders

(a) The Postal Service shall file the record in this appeal by December 26, 1996.

(b) The Secretary of the Postal Rate Commission shall publish this Notice and Order and Procedural Schedule in the Federal Register.

By the Commission.

Margaret P. Crenshaw,

Secretary.

Appendix

December 10, 1996, Filing of Appeal letter
December 13, 1996, Commission Notice and Order of Filing of Appeal
January 3, 1997, Last day of filing of petitions to intervene [see 39 C.F.R. § 3001.111(b)]
January 14, 1997, Petitioners— Participant Statement or Initial Brief [see 39 C.F.R. § 3001.115(a) and (b)]
February 3, 1997, Postal Service's Answering Brief [see 39 C.F.R. § 3001.115(c)]
February 18, 1997, Petitioners' Reply Brief should Petitioner choose to file one [see 39 C.F.R. § 3001.115(d)]
February 25, 1997, Deadline for motions by any party requesting oral argument. The Commission will schedule oral argument only when it is a necessary addition to the written filings [see 39 C.F.R. § 3001.116]
April 9, 1997, Expiration of the Commission's 120-day decisional schedule [see 39 U.S.C. § 404(b)(5)]

[FR Doc. 96–32097 Filed 12–17–96; 8:45 am]

BILLING CODE 7710–FW–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–38041; File No. SR–Phlx–96–11]

Self-Regulatory Organizations; Philadelphia Stock Exchange, Inc.; Order Granting Approval to Proposed Rule Change and Notice of Filing and Order Granting Accelerated Approval of Amendment No. 2 to Proposed Rule Change Relating to the Exchange's Calculation of Settlement Values for Cash/Spot Foreign Currency Option Contracts (''3–D Options'')

December 11, 1996.

On April 30, 1996, the Philadelphia Stock Exchange, Inc. (''Phlx'' or ''Exchange'') submitted to the Securities and Exchange Commission (''SEC'' or ''Commission''), pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] a proposed rule change to permit the Exchange to calculate settlement values for the cash/spot Dollar Denominated Delivery foreign currency option contracts (''3–D options'') and to limit the Exchange's liability in connection with the calculation and dissemination of these settlement values. On May 20, 1996, the Exchange submitted Amendment No. 1 to the proposed rule change.[3]

The proposed rule change, along with Amendment No. 1, was published for comment in the Federal Register on June 25, 1996.[4] On August 22, 1996, the Phlx clarified that it would not rely upon the proposed limitation of liability clause to limit the Exchange's liability for intentional misconduct or for any violation of the federal securities laws.[5] No comments were received on the proposal. This order approves the proposal, as amended by Amendment No. 2.

On March 8, 1994, the Commission approved trading for 3–D Foreign Currency Options on the Deutsche Mark.[6] Currently, the closing settlement value for 3–D options is calculated by a market information vendor acting as the Exchange's designated agent. The market information vendor will collect the bid and offer quotations for the current foreign exchange spot price from quotations submitted by at least fifteen interbank foreign exchange market participants, which the designated agent will select randomly from a list of twenty-five active interbank foreign exchange market participants. After discarding the five highest and the five lowest bids and offers, the market information vendor averages the

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

[2] 17 CFR 240.19b–4.

[3] *See* letter from Murray L. Ross, Vice President and Secretary, Phlx, to Anthony P. Pecora, Attorney, Division of Market Regulation, SEC, dated May 17, 1996 (''Amendment No. 1''). The changes contained in this letter were superseded by Amendment No. 2. *See infra* note 5.

[4] Securities Exchange Act Release No. 37323 (June 18, 1996) 61 FR 32880.

[5] *See* letter from Murray L. Ross, Vice President and Secretary, Phlx, to Anthony P. Pecora, Attorney, Division of Market Regulation, SEC, dated August 21, 1996 (''Amendment No. 2'') (superseding Amendment No. 1).

[6] Securities Exchange Act Release No. 33732 (Mar. 8, 1994), 59 FR 12023 (approving File No. SR–Phlx–93–10). Although the Commission has approved trading for 3–D Foreign Currency Options on the Japanese Yen, trading in these securities on the Exchange has not yet begun. Securities Exchange Act Release No. 36505 (Nov. 22, 1995), 60 FR 61277 (approving File No. SR–Phlx–95–42).