Tuesday, September 26

8:30–12:00 Role and functions of the NSRRC
1:15–3:30 Continued discussion on the role and functions of the NSRRC
3:30–4:30 Plans for subsequent meetings

Participants in parts of the discussion will include NRC staff as necessary.

Members of the public may file written statements regarding any matter to be discussed at the meeting. Members of the public may also make requests to speak at the meeting, but permission to speak will be determined by the Committee chairperson in accordance with procedures established by the Committee. A verbatim transcription will be made of the NSRRC meeting and a copy of the transcript will be placed in the NRC's Public Document Room in Washington, DC.

Any inquiries regarding this notice, any subsequent changes in the status and schedule of the meeting, the filing or written statements, requests to speak at the meeting, or for the transcript, may be made to the Designated Federal Officer, Dr. Jose Luis M. Cortez (telephone: 301–415–6596), between 8:15 am and 5:00 pm.

Dated at Rockville, Maryland this 24th day of August, 1995.

For the Nuclear Regulatory Commission.

Andrew L. Bates,

Federal Advisory Committee Management Officer.

[FR Doc. 95–21492 Filed 8–29–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 50–387 and 50–388]

Pennsylvania Power and Light Company; Correction

The March 29, 1995, **Federal Register** contained a ''Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing,'' for the Susquehanna Steam Electric Station. This notice corrects the notice published in the **Federal Register** on March 29, 1995, (60 FR 16192). The second sentence of the description section should read as follows: Specifically, for the refueling floor exhaust duct and wall exhaust duct radiation monitors, the proposed change would modify the applicable operational condition during specific control rod testing evolutions which are core alterations and would indicate that the operability requirement change does not apply during shutdown margin demonstrations.

Dated at Rockville, Maryland, this 22nd day of August 1995.

For the Nuclear Regulatory Commission.

Leonard N. Olshan,

Acting Director, Project Directorate I–2, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–21493 Filed 8–29–95; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 4, 1995, through August 18, 1995. The last biweekly notice was published on August 16, 1995 (60 FR 42597).

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

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

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

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

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

By September 29, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendments request: August 3, 1995

Description of amendments request: The proposed amendment changes would add the analytical method supplement entitled ''Fuel Rod Maximum Allowable Gas Pressure,'' CEN-372-P-A, dated May 1990, and its associated Nuclear Regulatory Commission Safety Evaluation Report, dated April 10, 1990, to the list of analytical methods in TS 6.9.1.10 used to determine the PVNGS core operating limits.

Basis for proposed no significant hazards consideration determination: As requied 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 does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident. The proposed change adds an NRC approved methodology and its associated Safety Evaluation Report (SER), to the list of analytical methods used to determine the core operating limits. The use of this methodology ensures that the consequences of an accident remain within the limits established by existing analyses. They do not alter any of the assumptions or bounding conditions currently in the UFSAR.

The U3C6 ECCS performance analysis included the analysis of the impact of the maximum calculated fuel rod gas pressures on the timing of cladding rupture and on the peak cladding temperature. This analysis concluded that the peak cladding temperature for Cycle 6 remained below that of the analysis of record and that the peak cladding temperature continued to occur at low burnup, specifically the burnup corresponding to the maximum initial fuel stored energy.

In addition to the LOCA analysis a DNB propagation analysis was performed to demonstrate that DNB propagation does not occur during postulated accidents that experience DNB when pressure in a fuel pin is higher than the system pressure. This analysis was performed using the fuel rod strain model described in CEN-372-P-A.

Based on these analyses, there is no increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed change. The intent of the proposed change is to utilize a new analytical method to ensure that the consequences of any equipment malfunction remain within the limits of existing analyses resulting in no impact on radiological consequences.

The impact of the maximum fuel rod gas pressures calculated for U3C6 was evaluated as part of the Cycle 6 ECCS performance analysis. Except for the highest burnup analyzed, the time of cladding rupture decreased as the initial fuel rod gas pressure increased with burnup. However, the peak cladding temperature occurred at the burnup with the maximum initial fuel stored energy. The analysis also determined that the ECCS performance analysis for U3C6 is bounded by that of the reference cycle analysis.

An evaluation was conducted to ensure that fuel would not experience DNB propagation when the pressure in a fuel pin is higher than the system pressure. DNB was shown not to propagate by demonstrating that the degree of cladding deformation is no more than the limit defined by the fuel rod maximum pressure Topical Report (CEN-372-P-A).

Therefore, it can be concluded that the proposed change to Section 6.9.1.10 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change adds an NRC approved Topical Report (methodology) and its associated SER, to the list of analytical methods used to determine core operating limits. The use of the new methodology ensures that safety margins are maintained within the results of existing calculations. Since the core operating limits will continue to be established by an NRC approved methodology and will provide adequate core protection, the proposed amendment does not involve a significant reduction in the margin of safety.

Analyses were conducted to determine the impact of higher fuel rod pressure on ECCS performance and DNB propagation. The results of the analyses show that the effects of higher fuel rod pressure are bounded by previous results.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involve no significant hazards consideration. Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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

NRC Project Director: William H. Bateman

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

Date of amendments request: July 13, 1995

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Technical Specifications (TSs) Section 5.2.1, ''Fuel Assemblies.'' The current TSs only allow fuel that is clad with either zircaloy or ZIRLO. The proposed change would allow the use of cladding material other than zircaloy or ZIRLO with an approved exemption. Thus, the proposed change will eliminate the need for future amendments to allow the use of different cladding material for which the Commission has issued an exemption.

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

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

Calvert Cliffs Technical Specification 5.2.1, Fuel Assemblies, states that fuel rods are clad with either zircaloy or ZIRLO. This reflects the requirements of 10 CFR 50.44, 50.46, and 10 CFR [Part] 50, Appendix K, which also restrict fuel rod cladding materials to zircaloy or ZIRLO. Baltimore Gas and Electric Company proposes to insert fuel assemblies into Calvert Cliffs Unit 1 which have some fuel rods clad in zirconium alloys that do not meet the definition of zircaloy or ZIRLO for testing purposes and has applied for an exemption to the regulations to allow that change. The proposed change to the Calvert Cliffs Technical Specifications will allow the use of cladding materials that are not zircaloy or ZIRLO with an approved exemption in accordance with 10 CFR 50.12.

The proposed change to the Unit 1 and Unit 2 Technical Specifications will allow the use of fuel rod cladding materials other than zircaloy or ZIRLO as long as those materials have been approved by an exemption to the regulations. To obtain approval of new cladding materials, 10 CFR 50.12 requires that the applicant show that the proposed exemption is authorized by law, is consistent with the common defense and security, will not present an undue risk to the public health and safety; and is accompanied by special circumstances.

Under the proposed change, any fuel rod cladding materials that are not zircaloy or ZIRLO must still be approved by the Nuclear Regulatory Commission (NRC) prior to use under 10 CFR 50.12. This change to the Technical Specifications allows the NRC to approve the use of cladding materials that are not either zircaloy or ZIRLO under 10 CFR 50.12 and not require an additional approval under 10 CFR 50.90. As such, the proposed change eliminates a duplicative regulatory requirement and would have no effect on the probability or consequences of an accident.

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

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

The proposed change eliminates a duplicated approval requirement and would have no effect on the possibility of a new or different type of accident. The proposed change to the Technical Specifications would allow the NRC to approve the use of fuel rod cladding materials that are not either zircaloy or ZIRLO under 10 CFR 50.12 and not require an additional approval under 10 CFR 50.90.

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

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

The proposed change eliminates a duplicated approval requirement and will have no effect on the margin of safety. The proposed change to the Technical Specifications would allow the NRC to approve the use of fuel rod cladding materials that are not either zircaloy or ZIRLO under 10 CFR 50.12, and not require an additional approval under 10 CFR 50.90.

Therefore, the proposed change does not involve a significant reduction in a margin 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 amendments request involves no significant hazards consideration.

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

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and

Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Ledyard B. Marsh

Commonwealth Edison Company, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Units 1 and 2, Will County, Illinois Docket Nos. STN 50-454 and STN 50-455, Byron Station, Units 1 and 2, Ogle County, Illinois Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station,Units 2 and 3, Grundy County, Illinois Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application for amendment requests: April 24, 1995

Description of amendment requests: The licensee proposes to amend Section 6 of the Technical Specifications of all ComEd stations to make the following changes: (1) delete the ‘‘Review, Investigative and Audit Functions’’ sections, in their entirety, and relocate these requirements to appropriate sections of the ComEd Quality Assurance Topical Report, (2) change titles to reflect the reorganization of ComEd’s Nuclear Operations Division, and (3) miscellaneous administrative and editorial changes.

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

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

(1) The proposed relocation of the ‘‘Review, Investigative and Audit Functions’’ sections of Technical Specifications to the QA Topical Report does not affect any accident initiators or precursors, and does not change or alter the design assumptions for the systems and components used to mitigate the consequences of an accident.

The relocation of these sections is consistent with the recommended changes specified in the October 25, 1993 letter from W. T. Russell (USNRC) to the Chairpersons of the Owner Groups’ Technical Specifications Committees, entitled, ‘‘Content of Standard Technical Specifications, Section 5.0, Administrative Controls’’.

Relocating these requirements to the QA Topical Report will continue to ensure that proposed future changes to these requirements will receive proper regulatory oversight. NRC review of the Quality Assurance Program is governed by 10CFR50.54. 10CFR50.54(a)(3) states: ‘‘Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC in accordance with the requirements of 50.71. Changes to the quality assurance program description that do reduce the commitments must be submitted to NRC and receive NRC approval prior to implementation, ...’’ Based on these 10CFR50.54 requirements, appropriate licensee and regulatory control of the requirements in the subject relocated Technical Specification sections will be maintained.

(2) The proposed title and organizational changes to Section 6 of Technical Specifications do not affect any accident initiators or precursors and do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident.

Commonwealth Edison’s organizational changes allow for increased senior management attention and oversight of station activities. Position titles and associated responsibilities have changed to increase the company’s efficiency in the management of its nuclear stations. These administrative changes do not reduce any requirements or commitments. The proposed changes enhance the administrative controls necessary to ensure safe plant operation.

(3) Other proposed administrative/editorial changes simply make corrections or provide needed clarification prompted by the reorganization. These changes provide consistency with station procedures, programs, other Technical Specifications, and Standard Technical Specifications. They are administrative in nature and do not impact any accident previously evaluated in the UFSAR.

In conclusion, none of the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated.

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

(1) The proposed relocation of the ‘‘Review, Investigative and Audit Functions’’ sections of Technical Specifications to the QA Topical Report does not affect the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation and no new or different type of equipment will be installed that could give rise to a new or different kind of accident that was previously evaluated.

The proposed changes are considered to be administrative or programmatic in nature and do not affect equipment or components that could initiate an accident. All administrative commitments being relocated to the QA Topical Report will continue to receive appropriate regulatory oversight pursuant to 10CFR50.54.

(2) The proposed title and organization changes do not affect the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation; no new or different type of equipment will be installed. The proposed changes are considered to be administrative changes that will enhance the performance of organizations responsible for the safe operation of the plant to respond to plant transients or emergencies. All responsibilities described in Technical Specifications for management activities will continue to be performed by qualified individuals.

(3) All other proposed changes are administrative in nature and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

(1) The proposed changes are administrative or programmatic in nature and do not affect the margin of safety for any safety parameters and setpoints addressed in Technical Specifications. The assumptions, initial conditions and methodologies used in the accident analyses remain unchanged, therefore, accident analyses results are not impacted.

Placing these requirements in QA Topical Report will continue to ensure that proposed future changes to these requirements will receive proper regulatory oversight pursuant to 10CFR50.54.

(2) The proposed title and organizational changes are administrative in nature and do not affect the margin of safety for any Technical Specification. The initial conditions and methodologies used in the accident analyses remain unchanged, therefore, accident analyses results are not impacted.

(3) All other proposed changes are administrative in nature and have no impact on the margin of safety for any Technical Specification.

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

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

Local Public Document Room location: for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481; for Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for LaSalle, Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021; for Zion, 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 60690

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: June 8, 1995

Description of amendment request: The proposed amendments would revise Technical Specifications Section 3/4.8, Electrical Power Systems, and the associated Bases for LaSalle County, Byron, and Braidwood Stations. The proposed changes revise surveillance and administrative requirements associated with emergency diesel generators (EDGs) in accordance with the guidance of NRC Generic Letter 94-01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' and Regulatory Guide (RG) 1.9, ''Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants.'' The proposed changes include: (1) eliminating increased testing requirements for EDGs, (2) eliminating special reporting requirements for EDGs, (3) eliminating the semi-annual fast load test and replacing it with a requirement to load EDGs semi-annually in accordance with the vendor recommendations for all test purposes other than the refueling outage Loss of Offsite Power (LOOP) tests, (4) de-coupling the 24-hour endurance run and the LOOP/loss-of-coolant (LOCA) (LOOP only for LaSalle) sequencing requirements for the hot start test, (5) removing RG 1.108 references to testing requirements, (6) eliminating testing requirements when an EDG becomes inoperable due to an inoperable support system, an independently testable component, or preplanned maintenance or testing, or if there is not a potential common mode failure for the remaining diesel generator, (7) deleting the requirement for inspecting the EDGs in accordance with procedures prepared in conjunction with its manufacturer's recommendations, and (8) making editorial changes.

Basis for proposed no significant hazards consideration determination: As requied 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:

The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions affecting the ability of the EDGs to mitigate the consequences of an accident.

Deleting the special reporting requirements from the Technical Specifications is administrative. ComEd will continue to notify the Commission of significant EDG failures in accordance with 10 CFR 50.72 and 50.73 criteria.

Excessive testing requirements have proven to be a contributor to increased equipment degradation. Removing inappropriate and redundant requirements increases EDG reliability and enhances the ability of EDGs to mitigate the consequences of an accident. Implementing ComEd's alternative to the maintenance rule for the EDGs provides additional assurance that high EDG performance will be maintained.

EDG equipment degradation will be reduced by eliminating the semi-annual fast load test for EDGs in accordance with the vendor recommendations for test purposes other than the refueling outage Loss of Offsite Power (LOOP) tests. This improves EDG reliability and availability and further enhances their ability to mitigate the consequences of an accident. The LOOP test would still be performed to provide assurance that the EDG is capable of responding to a LOOP as assumed in the accident analyses.

De-coupling the 24 hour endurance test and the LOOP/LOCA (for LaSalle, LOOP) sequencing test requirements for the hot start test has no effect on accident mitigation. Demonstrating diesel generator hot restart capability without loading the engine does not invalidate or reduce the effectiveness of the hot restart test. The hot restart test can be conducted in any plant condition since its performance at power will have no adverse effect on plant operations.

The proposed editorial changes are administrative in nature. They improve readability and provide consistency with current industry guidance.

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

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

The proposed changes do not alter the ability of the EDGs to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in plant safety analyses. The proposed changes have no impact on component or system interactions, or the plant design basis.

Instrumentation setpoints, starting, sequencing, and loading functions associated with EDGs are not affected by the proposed changes. Furthermore, combining the alternate EDG system maintenance rule implementation program with the proposed amendment will enhance both the availability and the performance of the EDGS.

Therefore, there is not a potential for creating the possibility of a new or different type of accident from any accident previously evaluated.

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

The proposed changes do not increase the probability or consequences of an accident, and there is no impact on equipment design or operation. The proposed changes do not affect the results of accident and transient analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of safety analyses. There is no associated change to the type, amount, or control of radioactive effluents, nor is there an associated increase in individual or cumulative occupational radiation exposure. There is no effect upon the capabilities of the associated systems to perform their intended functions within the allowed response times assumed in safety analyses.

The proposed changes are compatible with plant operating experience and are consistent with the guidance provided in NUREG-1366, Generic Letters 93-05 and 94-01, and Regulatory Guide 1.9. In two instances ComEd's proposed changes deviate from these guidance documents. However, the changes are consistent with the intent of the documents or other NRC guidance documents. Eliminating excessive testing requirements can improve safety by reducing challenges to plant systems and reducing equipment wear and degradation. While the proposed changes affect surveillance intervals; there are no changes to the methods used to perform the surveillances.

EDG reliability and availability will be improved by the proposed changes. The surveillances will continue to demonstrate the ability of the EDGs to perform their intended function of providing electrical power to the emergency safety systems needed to mitigate design basis transients. No margin of safety is reduced.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the standard review plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. The proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification.

Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), ComEd has concluded that the proposed change does not constitute a significant hazards consideration.

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

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481; for LaSalle, 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. 50-237 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 requests: August 30, 1994, as supplemented August 4, 1995.

Description of amendment requests: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operation and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The August 30, 1994, and August 4, 1995, applications proposed to upgrade only Section 3/4.2 (Instrumentation) of the Dresden and Quad Cities TS.

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

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes to the current Technical Specifications (CTS) represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.2 are based on BWR-STS (NUREG-0123, Revision 4 ''Standard Technical Specifications General Electric Plants BWR/4) guidance or NRC accepted changes at later operating BWR plants. Any deviations from BWR-STS and CTS requirements do not significantly increase the probability or consequences of any previously evaluated accident for Dresden and Quad Cities Station. These proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Instrumentation Systems are not assumed in any safety analysis to initiate any accident sequence for both Dresden and Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. These more prescriptive surveillance requirements increase the probability that the Instrumentation Systems will perform their intended functions. Therefore, the proposed TS will improve the reliability and availability of all affected systems and reduce the consequences of any accident previously evaluated.

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

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station, other than technically valid trip setpoint changes. Some of the changes may involve revision in the operation of the station; however, these changes provide additional restrictions which are in accordance with the current safety analyses, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.2 is based on BWR-STS guidelines or NRC accepted changes at later operating BWR plants. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the BWR-STS or later operating BWRs. Any deviations from BWR-STS or CTS requirements do not create the possibility of a new or different kind of accident than previously evaluated for Dresden and Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Various surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications or adopt requirements that have been used at other operating BWRs with designs similar to Dresden and Quad Cities. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Instrumentation Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Instrumentation Systems are generally more prescriptive than the current requirements specified within the Technical Specifications; 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:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.2 implements present requirements in accordance with the guidelines set forth in the BWR-STS. Any deviations from BWR-STS and CTS requirements do not significantly reduce the margin of safety for Dresden and Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden and Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and Quad Cities and maintain necessary levels of system or component readability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Instrumentation Systems when required to mitigate accident conditions; 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 requests involve no significant hazards consideration.

Local Public Document Room location: for Dresden, Morris Public Library, 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 60690

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: March 8, 1995, as supplemented June 1, 1995

Description of amendment request: The proposed amendments would revise the secondary undervoltage setpoint.

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

1. The proposed amendment does not involve an increase in the probability of occurrence or consequences of any accident previously evaluated.

The proposed amendment does not change the fundamental function or capability of the Secondary Undervoltage protection as described in UFSAR section 8.3. Inadvertent or spurious operation of the Secondary Undervoltage protection function will initiate loading of the safe shutdown loads on the diesel generators and is not assumed to initiate an accident. The proposed Secondary Undervoltage setpoints are low enough to prevent spurious actuations given the expected off site grid voltages.

This change does not affect the initiators or precursors of any accident previously evaluated. This change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. The change in setpoints for the Secondary Undervoltage protection system does involve some changes to existing plant equipment (such as transformer tap changes and Circulating Water pump excitation circuit changes). However, all changes to existing plant equipment have been or will be evaluated in accordance with the requirements of 10CFR50.59 prior to installation, to determine that no unreviewed safety questions exist with regard to the plant changes.

Since any design changes have been or will be determined to be acceptable per 10CFR50.59 prior to installation and no new plant equipment will be installed, the probability of occurrence of accidents previously evaluated will not increase.

With Zion Station's new Auxiliary Power System configuration and the proposed Secondary Undervoltage setpoints, the probability of a Loss of Off-Site Power (LOOP) is actually reduced since the original Auxiliary Power System configuration and Secondary Undervoltage setpoints required a higher grid voltage to ensure that safety related loads would be powered from Off-Site power sources during a design basis accident.

The consequences of accidents previously evaluated are not increased. The proposed change does not affect the required level of availability or systems required to mitigate the accidents considered in the Analyses. Administrative controls will be in place to ensure that the installed setpoints are low enough to ensure that the Emergency Diesel Generators are not unnecessarily challenged. The proposed changes will increase the level of confidence that the ESF equipment will be capable of starting and operating during a design basis accident with degraded off-site grid voltage. The increase in the level of confidence is the result of the more rigorous methodology used to determine limited ESF bus voltages, given the minimum expected off-site AC voltage. Based on the previous discussion, it is determined that there will be no significant increase in the consequences of any accident previously evaluated.

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

The proposed Secondary Undervoltage setpoint change does not change the design of the Secondary Undervoltage protection system or its function to protect against degraded offsite power. Actuation of the Secondary Undervoltage protection system will initiate a sequence of events that will start the Emergency Diesel Generator (EDG) for the associated ESF bus, strip all loads from the bus, open all feed breakers to the bus, close the Emergency feed breaker (thus energizing the bus from the EDG), and initiate sequenced starting of the Safe Shutdown equipment supplied by the bus, including a Service Water pump, Component Cooling Water pump, Auxiliary Feedwater pump, and Reactor Containment Fan Cooler(s), as applicable.

The proposed change does not involve the addition of any new or different types of equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the Final Safety Analysis Report. No safety related equipment or function will be altered as a result of this proposed change. Because no new failure modes are introduced, the proposed amendment does not create a new or different kind of accident from any previously analyzed in the UFSAR.

Based on the above discussion, the proposed amendment does not create a new or different kind of accident from any previously analyzed in the UFSAR.

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

The proposed amendment will allow the Secondary Undervoltage setpoint to be conservatively established based on new engineering calculations which consider the lowest expected offsite grid voltage and operation of all required ESF equipment under design basis accident loading conditions.

The proposed Secondary Undervoltage setpoints will provide increased confidence that adequate bus voltage will be available to support starting and operation of all required ESF loads. The proposed setpoint includes worst case instrument error to ensure that the lowest possible voltage will not be lower than the degraded voltage analytical limits. Additionally, the proposed setpoints are low enough to prevent spurious actuations due to expected fluctuations in the grid voltage. The new setpoints are based on a minimum expected grid voltage of 343 kV, with added margin. The proposed changes will provide an increase in the level of protection that currently exists and will ensure the margin of safety is adequately maintained.

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

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: August 3, 1995

Description of amendment request: The proposed amendment will add an one-time footnote to Technical Specification (TS) Section 3/4.7.12, ''Ultimate Heat Sink,'' to increase the allowed outage time from 6 hours to 18 hours for the months of August and September. In addition, also for the months of August and September, the maximum service water limit will be elevated from 90°F to 95°F.

Basis for proposed no significant hazards consideration determination: As requied 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 analyzed.

The proposed addition of a 12 hour time period to monitor the ultimate heat sink temperature to the Technical Specification Limiting Condition for Operation action statements does not involve an increase in the probability of an accident previously evaluated. The probability of an accident previously evaluated is not increased by a short-term increase in the ultimate heat sink temperature. An evaluation has been performed that safe shutdown will be achieved and maintained for a loss of normal AC power event with the additional consideration of a single failure with service water inlet temperatures as high as 95°F. In addition, an evaluation of the credible FSAR Chapter 15 events with AC power available and no isolation of non-essential service water loads has been performed that demonstrates that safe shutdown will be achieved and maintained. There has been no significant increase in the consequences of these events previously evaluated.

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

The proposed technical specification change does not create the possibility of a new or different kind of accident previously analyzed. The addition of a 12 hour time period to monitor the ultimate heat sink temperature increases the amount of time that is allowed for the plant to be in Hot Standby from 6 to 18 hours should the ultimate heat sink temperature increase above 90°F. This extension of the time allowed for the plant to be in Hot Standby does not change the plant configuration. As such, the change does not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed technical specification change does not involve a significant reduction in the margin of safety. The addition of a 12 hour time period to monitor the ultimate heat sink temperature increases the time required for the plant to be in Hot Standby from 6 to 18 hours should the ultimate heat sink temperature exceed 90°F. An evaluation has been performed to demonstrate that the risk significance associated with the increased action time is very low. In addition, safe shutdown capability has been demonstrated for service water inlet temperatures as high as 95°F.

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: Russell Library, 123 Broad Street, Middletown, CT 06457

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 5, 1995

Description of amendment request: The proposed amendment would change the surveillance frequency of radiation area, and effluent and process monitors from monthly to quarterly; and the required frequency for minimum exercise of control element assemblies also from monthly to quarterly.

Basis for proposed no significant hazards consideration determination: As requied 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 staff's review is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Extending surveillance test intervals as proposed will reduce the probability of inadvertent reactor scrams and ensuing challenges to safety systems. This is accomplished by reducing the occasions and thus the total time that the subject systems are removed from their ''normal'' configuration and placed into the required ''test'' configuration. In addition, the probability of test-induced failures, or failures caused by human error, is likewise decreased. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Extending surveillance test intervals as proposed will not require installation of any new or different equipment, and will not alter or otherwise modify existing plant equipment. Thus, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Independent research has found that equipment failures and personnel errors during several types of surveillance tests caused a significant number of reactor scrams and attendant unnecessary challenges to safety equipment. The results of this research have been corroborated by the licensee's plant specific operating experience. The licensee concludes that the reduced test intervals proposed in this amendment remain sufficient to ensure known phenomena, such as instrument setpoint drift and random hidden failures, remain within the assumptions of the safety analysis. Thus, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011

NRC Project Director: Phillip F. McKee

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

Date of amendment request: July 24, 1995.

Description of amendment request: The proposed amendment would delete Table 3.4-1, ''Reactor Coolant System Pressure Isolation Valves'' from the Seabrook Station, Unit No. 1 Technical Specification section 3.4.6.2. Reference to Table 3.4-1 also would be deleted from Limiting Condition for Operation 3.4.6.2 f and from Surveillance Requirement 4.4.6.2.2. The information contained in Table 3.4-1 would be relocated to the Technical Requirements Manual.

Basis for proposed no significant hazards consideration determination: As requied 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 proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because they do not in any way alter the operability or surveillance requirements for pressure isolation valves. The proposed changes merely delete a listing of valves which are designated as pressure isolation valves in accordance with the definition provided in 10 CFR Part 50. Therefore, neither the probability nor consequences of previously evaluated accidents are affected.

B. The proposed 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 they do not affect in any way the manner by which the facility is operated or make any changes in structures, systems, or components which could affect the operational characteristics of the facility.

C. The proposed changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not affect the operability requirements or surveillance testing of any pressure isolation valve and do not affect in any way the manner by which the facility is operated or involve equipment or features which affect the operational characteristics of the facility.

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: Phillip F. McKee

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

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment adds Technical Specifications (TS) to Section 3.10, Refueling and Spent Fuel Handling. Specifically, the proposed TS (with applicability, action, and surveillance requirements) will require that: (1) the reactor be subcritical for at least 100 hours before the start of reactor refueling operations, (2) the spent fuel pool bulk temperature be maintained less than or equal to 140°F, and (3) two trains of shutdown cooling be operable during reactor refueling operations. In support of the request, NNECO proposes to: (1) use the ORIGEN2 code to more accurately predict decay heat loads from the spent fuel, (2) use the ONEPOOL code to credit the effect of evaporative cooling on the spent fuel pool bulk temperature, and (3) take credit for both trains of shutdown cooling to assist the spent fuel pool cooling system during refueling outages. In addition, the proposed amendment modifies the table of contents and associated Bases section to reflect the changes.

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

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The proposed license amendment will allow NNECO to use the shutdown cooling system (SCS) to assist the spent fuel pool cooling (SFPC) system to cool the spent fuel pool during refueling outages. This amendment request does not affect: the number of spent fuel bundles allowed in the spent fuel pool, spent fuel pool criticality analysis, structural analysis of the spent fuel pool, or radiological release scenarios.

The proposed license amendment also allows NNECO to use ORIGEN2 and ONEPOOL codes. The ORIGEN2 code more accurately predicts decay heat loads from the spent fuel in the spent fuel pool. The ONEPOOL code credits the effect of evaporative cooling on the spent fuel pool bulk temperature. The use of these codes will improve the accuracy of predicting spent fuel pool bulk temperatures during normal and abnormal refueling scenarios.

The use of the SCS to assist the SFPC system to cool the spent fuel pool will allow the movement of spent fuel to begin 100 hours after reactor shutdown. The existing accident analysis for a dropped spent fuel bundle during refueling bounds this situation as the analysis assumed a decay time of 24 hours.

The three new proposed technical specifications will provide sufficient controls on the movement of spent fuel into the spent fuel pool, bulk temperature of the spent fuel pool and operability of the shutdown cooling system to operate within analysis assumptions during refueling operations at Millstone Unit No. 1.

Therefore, based on the above, the use of the SCS to assist the SFPC system to cool the spent fuel pool during refueling outages, the use of the ORIGEN2 code, the use of the ONEPOOL code, and the addition of three technical specifications will not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed license amendment to use the SCS to assist the SFPC system to cool the spent fuel pool will allow SCS train B to cool the spent fuel pool in a method similar to train A.

The proposed license amendment to use ORIGEN2 and ONEPOOL codes to predict spent fuel pool bulk temperatures will increase the accuracy of analyzing normal and abnormal refueling scenarios.

The three new proposed technical specifications will sufficiently control refueling operations to support analyzed accident scenarios.

Therefore, the use of the SCS to assist the SFPC system to cool the spent fuel pool, the use of the ORIGEN2 code, the use of ONEPOOL code and the addition of three technical specifications do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed license amendment to use the SCS to assist the SFPC system to cool the spent fuel pool will allow the crediting of the SCS and SFPC system to remove heat from the spent fuel pool during normal refueling scenarios. The analysis demonstrates that this cooling configuration will maintain the spent fuel pool bulk temperature below the pool design limit of 140°F with a postulated single active failure.

The addition of the train B SCS cross-tie does not adversely affect the existing design basis of the SCS to remove sensible and decay heat from the reactor water, cool it from 280°F to 125°F within 24 hours, and to maintain the reactor water at 125°F.

The proposed license amendment to use ORIGEN2 and ONEPOOL codes will improve the accuracy of predicting spent fuel pool bulk temperatures during normal and abnormal refueling scenarios.

The thermal hydraulic analysis most limiting time to boil calculation of 5.4 hours for loss of all forced cooling to the spent fuel pool is consistent with assumed operator response times for similar scenarios.

The three new proposed technical specifications will ensure that the margin of safety established by engineering analysis of refueling operations is maintained.

Therefore, based on the above, the use of the SCS to assist the SFPC system to cool the spent fuel pool, the use of the ORIGEN2 code, the use of the ONEPOOL code, and the addition of three technical specifications does not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket Nos. 50-245, 50-336 and 50-423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London, Connecticut

Date of amendment request: August 4, 1995

Description of amendment request: The proposed license amendments will modify the Administrative Controls Section (Section 6) of the Millstone Unit Nos. 1, 2, and 3 Technical Specifications to allow the Plant Operations Review Committee (PORC) and Site Operations Review Committee (SORC) to direct its efforts in the review of more critical safety matters which affect day-to-day operation. This will be accomplished by the establishment of a Station Qualified Reviewer Program (SQRP) and the reassignment of certain procedure approvals to designated managers in lieu of approval by PORC/SORC.

Basis for proposed no significant hazards consideration determination: As requied 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:

...These proposed changes do not involve an SHC because the changes do not:

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

These changes are administrative in nature. They do not involve any modifications to plant systems and do not alter the method of operation of any plant equipment. The change involves the establishment of a SQRP for the review of plant procedures, programs or changes thereto that do not involve a 10CFR50.59 evaluation.

Implementing a SQRP will not result in a degradation of the current level of procedure review. PORC/SORC will retain the responsibility for reviewing any document for which a 10CFR50.59 evaluation is required. Personnel selected to be SQRs [Station Qualified Reviewers] will possess the technical experience and expertise to provide a thorough technical review as required by plant procedures. These personnel, and the managers authorized to approve these procedures, will be designated in writing by the Unit Director or the Senior Vice President - Millstone Station. Procedures or classes of procedures that can be reviewed per the SQRP will be specified in writing by the Unit Director or the Senior Vice President - Millstone Station. Procedures will receive an appropriate cross-disciplinary review when necessary.

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

The proposed technical specification changes do not change the design or function of any plant structure, system, or component, nor do they introduce any new failure modes. As stated above, the implementation of a SQRP will not degrade the quality of plant procedures.

There are no modifications to plant structures, systems, or components associated with these proposed changes, and the operation of plant equipment and systems remain unchanged. Since the changes proposed in this license amendment request do not revise existing plant structures, systems, or components, do not change the manner in which the plant is operated and, do not change the manner in which the plant will respond to any design basis accidents, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The changes proposed in this proposed license amendment request do not affect the ability of any system to perform its safety-related function. As described above, these proposed changes are administrative in nature. They do not change any plant operating parameters or design features and do not reduce the level of effectiveness of any existing administrative controls. The proposed change will not result in changes to the bases for any technical specification. The establishment of the SQRP will continue to provide for the adequate review of procedures. In addition, another direct benefit of this program is that the amount of material presented to PORC/SORC will decrease. The reduction in the amount of material presented to PORC/SORC for review will allow the PORC/SORC to focus on safety significant issues. Since none of the assumptions in the technical specifications bases are affected by the changes presented in this license amendment request, the margin of safety which exists in the current technical specifications 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: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 22, 1995

Description of amendment request: The proposed changes modify the facility requirements for thermal-hydraulic instability avoidance and protection to address concerns over reactor fuel performance during instability events. Changes are proposed to the Technical Specifications to utilize the flow biased Average Power Range Monitor high neutron flux scram and a power-flow map exclusion region consistent with one of the NRC approved BWR Owners' Group solutions. In addition, a change to correct an error in the Average Planar Linear Heat Generation Rate during single loop operation is also proposed.

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

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

The implementation of BWR Owner's Group long term stability solution Option 1-D at Monticello does not modify the assumptions contained in the existing accident analysis. The use of an exclusion region and the operator actions required to avoid and minimize operation inside the region do not increase the possibility of an accident. Conditions of operation outside of the exclusion region are within the analytical envelope of the existing safety analysis. The operator action requirement to exit the exclusion region upon entry minimizes the probability of an oscillation occurring. The actions to drive control rods and/or to increase recirculation flow to exit the region are maneuvers within the envelope of normal plant evolutions. The flow based scram has been analyzed and will provide automatic fuel protection in the event of a core wide instability. Thus, each proposed operating requirement provides defense in depth for protection from an instability event while maintaining the existing assumptions of the accident analysis. The proposed change to the method by which the MAPLHGR [maximum average planar linear heat-generation rate] is obtained for single loop operation is consistent with the analysis performed for the Average Power Range Monitor/Rod Block Monitor Technical Specifications (ARTS) program. The analysis performed in support of the ARTS program demonstrated that the limits established assure compliance with fuel limits. Therefore, this amendment will not cause a significant increase in the probability or consequences of an accident previously evaluated for the Monticello plant.

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

As stated above, the proposed operating requirements either mandate operation within the envelope of existing plant operating conditions or force specific operating maneuvers within those carried out in normal operation. Since operation of the plant with all of the proposed requirements is within the existing operating basis, an unanalyzed accident will not be created through implementation of the proposed change. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident.

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

Each of the proposed requirements for the plant thermal-hydraulic stability provides a means for fuel protection. The combination of avoiding possible unstable conditions and the automatic flow biased reactor scram provides an in-depth means for fuel protection. Therefore, the individual or combination of means to avoid and suppress an instability supplements the margin of safety. The operating limits established for the single loop operation MAPLHGR provide an acceptable margin of safety as demonstrated in NEDC-30492, ''Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant-April 1984.'' The proposed amendment will not involve a reduction in the margin of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: July 5, 1995

Description of amendment request: The proposed amendment, part of the Monticello Surveillance Test Interval/Allowed Outage Time (STI/AOT) Program, extends the surveillance test intervals and allowable out-of-service times for selected instrumentation. The proposed changes are intended to minimize unnecessary testing and remove excessively restrictive out-of-service times that could potentially degrade overall plant safety and availability.

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

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

The maximum failure frequency change is for the ECCS Actuation Instrumentation as identified by General Electric topical report NEDC-30936P-A, and Monticello specific report RE-006. These reports concluded core damage frequency changed by less than 4% when STIs were increased to once per 3 months, AOTs for surveillance were increased to 6 hours, and AOTs for repair were increased to 24 hours. Since this small increase was within the guideline of acceptability stated in NEDC-30936P-A, and Monticello only proposes to increase the repair AOT to 12 hours rather than 24 hours, this amendment will not cause a significant increase in the probability or consequences of an accident previously evaluated for the Monticello plant (see RE-006).

The drift analysis determined the associated instrumentation would not be adversely effected with the longer calibration intervals. Pertinent process parameters including instrument drift will still be within acceptance criteria with the longer surveillance intervals.

The recirculation flow meters and flow instrumentation are not used in any safety or accident analysis. Therefore, no analysis would be changed by increasing the calibration interval to once per cycle.

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

These changes only affect the instrument STI and AOT times. No changes are being made to the functions of the instrumentation. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident.

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

These changes will improve the performance of equipment and are intended to reduce the potential for equipment failures due to unnecessary testing. The safety limits and the limiting safety system setpoints will not be affected by these changes. No safety margins are affected, therefore, the drift will remain within the margins of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: August 4, 1995

Description of amendment request: This proposed amendment would revise the Technical Specifications (TS) for the requirements for the containment radiation high signal (CRHS) and the safety injection and refueling water (SIRW) tank low signal (STLS) contained in TS 2.15, Tables 2-3 and 2-4. Specification 3.1, Table 3-2 will also be revised to include administrative changes to the CRHS surveillance methods to be consistent with the applicable surveillance functions. The Basis for Specification 2.15 is being revised to clarify that the number of installed channels for CRHS is two. The term ''SOURCE CHECK'' is being deleted from the Definitions section.

Basis for proposed no significant hazards consideration determination: As requied 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.

The Omaha Public Power District (OPPD) proposes to revise Technical Specification (TS) 2.15, Table 2-3 by revising the requirement for placing the Safety Injection Refueling Water (SIRW) tank low level channel(s) in the tripped condition to placing them in the bypassed condition. Due to the derived signal, if a channel was in the tripped condition and a single failure occurred, (that being one channel of STLS on either A or B circuits), a premature SIRW tank low signal (STLS) would be generated. During a design basis accident (DBA) with a valid Containment Pressure High Signal (CPHS) or Pressurizer Pressure Low Signal (PPLS), this single failure would prevent the contents of the SIRW tank from being injected into the reactor coolant system. The resulting logic of placing the SIRW tank low level channels in BYPASS rather than TRIP would not cause a premature switchover of the high pressure safety injection pumps to the containment sump and it would not prevent the switchover when needed.

OPPD also proposes to revise TS 2.15, Table 2-4, by reducing the number of minimum operable Containment Radiation High Signal (CRHS) channels from two to one. This proposed change revises the requirements of TS 2.15 to coincide with changes to the TS and Offsite Dose Calculation Manual (ODCM) that were implemented by TS Amendment 152. The Engineered Safety Feature (ESF) actuation system supervisory A and B safeguard initiation channels will not be affected by this proposed TS change. The minimum level of engineered safeguards performance acceptable for the DBA, (i.e., minimum safeguards) will continue to be maintained in accordance with IEEE 279 - 1971, ''Criteria for Protection Systems for Nuclear Power Generating Stations.''

Included in this change are administrative revisions to TS 3.1, Table 3-2, for replacing the current surveillance methods for checking and testing the CRHS instrumentation with the defined terms ''CHANNEL CHECK'' and ''CHANNEL FUNCTIONAL TEST,'' respectively. These proposed revisions are administrative in nature and reflect TS-defined terminology for the instrumentation surveillance methods utilized to ensure that the CRHS instrumentation is operable. A channel check requires a qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. A channel functional test requires the injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating actions. Other proposed administrative changes include deleting the term ''SOURCE CHECK'' from the TS Definitions section as source check will no longer be used in the FCS TS and adding verbiage to the TS 2.15 Basis for clarifying that the number of installed channels for CRHS is two.

Therefore, the proposed change, as described above, would not increase 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.

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of the proposed changes to TS 2.15, Tables 2-3 and 2-4. The proposed revisions to TS 3.1, Table 3-2 are administrative changes to make the TS more accurately reflect defined terminology and the methods utilized to ensure that the CRHS instrumentation is operable. The proposed TS revisions do not require any changes to the present methods of verifying CRHS instrumentation operability. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

There are no changes to the equipment or plant operations as a result of the changes being made to the number of minimum operable CRHS channels. The proposed changes to the STLS will require that the inoperable channel be placed in BYPASS rather than TRIP. This action would ensure that a single failure would not cause a premature safety injection switchover to the containment sump and would not prevent switchover when needed. Therefore, this proposed change does not reduce a margin of safety.

The proposed revisions to TS 3.1, Table 3-2 are administrative changes to make the TS more accurately reflect defined terminology and the methods utilized to ensure that the CRHS instrumentation is operable. The proposed TS revisions do not require any changes to the present methods of verifying CRHS instrumentation operability. The proposed changes to the Definitions and TS 2.15 Basis sections are administrative in nature. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: June 22, 1995

Description of amendment request: The amendments would revise the Technical Specifications 3.4.1.4 and 3.9.8.2 by deleting footnotes and associated information regarding Service Water header operation and its support function for Residual Heat Removal operation. These footnotes and associated information had been placed in the Technical Specifications because of the concern about Service Water system piping integrity in the mid-1980's.

Basis for proposed no significant hazards consideration determination: As requied 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. Do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Even though one service water loop will be out for maintenance, both loops of residual heat removal (RHR) will be kept operable, consistent with the requirements of STS (NUREG 1431). A minimum of two RHR, two component cooling (CC), and two service water (SW) pumps, powered from two different vital busses, will be kept operable.

Only one component cooling heat exchanger will be operable since only one service water loop is operable. The CC heat exchangers for both Units 1 and 2 have a very high reliability. The primary heat transfer surfaces of the heat exchangers are made of titanium; no material problems have been experienced in ten years of service.

The remaining active components that, through misoperation, could potentially defeat RHR capability are, (1) the motor operated valves in RHR or SW that could develop a ''hot short'' and subsequently close and (2) the air operated temperature/ flow control valves of the CC heat exchangers. Additional actions will be taken to effectively eliminate the possibility of these single point valves from failing and defeating RHR capability. The motor operator breakers will be tagged open during MODES 5 and 6, except for flooding the cavity, when the RHR suction valves must be closed. The CC Heat Exchanger air operated temperature/flow control valves fail open, or as is, on loss of air which is the safe position. Operators will monitor critical temperatures; this equipment is accessible if any corrective action is required. Thus, with one service water header out of service, the intent of the technical specifications as defined in the bases section (to have a single failure proof RHR system) is met with the proposed system configuration. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The catastrophic failure of a moderate energy Class 3 piping system is not a credible event, based on the upgraded reliability of the system, the redundancy of active components, the elimination of single failure points, and on the industry and regulatory positions established for this type of system. Since SW is a Class 3 moderate energy system, the only postulated passive failure mode is a leakage crack. In accordance with Generic Letter (GL) 91-18 and GL 90-05, a leak in the SW system, following acceptable evaluation, does not constitute a failure that causes the loss of capability to perform it's intended safety function. A moderate energy Class 3 piping leak does not cause the system to be declared inoperable. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any previously evaluated.

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

RHR redundancy is maintained; no credible single failure point exists that could cause a nonrecoverable loss of SW. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

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

NRC Project Director: John F. Stolz

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: September 15, 1992, as supplemented April 20, 1993, April 26, 1995, and July 27, 1995.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3.1.1.4, 3.1.1.6, and 4.3.4, and add a Basis to address Generic Letter (GL) 90-06. GL 90-06 represents the technical resolution of Generic Issue (GI) 70, ''Power Operated Relief Valve and Block Valve Reliability,'' and GI 94, ''Additional Low Temperature Overpressure Protection for Light Water Reactors.'' The resolution of these issues proposes new requirements and TS changes that enhance the reliability of power-operated relief valves (PORVs) and block valves along with TS changes that will provide additional low-temperature overpressure protection (LTOP).

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

There is no significant increase in the probability or consequences of an accident previously evaluated because the accident conditions and assumptions are not significantly affected by the proposed change.

The proposed change to action statement 3.1.1.4a(i) [proposed to be renumbered to 3.1.1.6c] to include the removal of power from a closed block valve will provide additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be operable to assure RCS [reactor coolant system] integrity.

The provision of the generic letter requires, with one or both PORV(s) inoperable to initiate shutdown actions if PORV operability is not restored within 72 hours or 1 hour respectively. RG&E [Rochester Gas and Electric Corporation] does not address these shutdown actions, but rather will concentrate on re-establishing valve operability. If the block valve(s) and power are not removed within 1 hour shutdown provisions must be initiated. [***].

Proposed action statement 3.1.1.4a(ii) [proposed to be renumbered to 3.1.1.6d] includes a provision to place the block valves associated PORV(s) switch in manual control due to an inoperable block valve(s). This requirement precludes the automatic opening for an overpressure event to avoid the potential for a stuck-open PORV at a time that the block valve is open and inoperable. [***].

The proposed change of maintaining power to closed block valves could potentially increase the probability of an inadvertent opening of a block valve. The safety impact is, however, not significant since the proposed changes are only applicable if the PORV is inoperable due to excessive seat leakage (proposed action 3.1.1.6b). [***].

Proposed action statement 3.1.1.6b establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage and is therefore considered operable to perform its intended safety function. [***].

Proposed Surveillance Requirement 4.3.4.3 addresses operability of the Nitrogen System by demonstration of the PORVs at least once per 18 months by operating the PORVs through a complete cycle of full travel. [***].

Based on the above efforts, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The possibility of a new or different kind of accident from any previously evaluated is not created. In matters related to nuclear safety, all accidents continue to bound previous analyses. The proposed changes do not add or modify any equipment design nor do the proposed changes involve any significant operational changes to any plant systems.

The proposed amendment does not involve a significant reduction in the margin of safety as defined in the basis for any technical specification because the results of the accident analyses which are documented in the UFSAR [Updated Final Safety Analysis Report] continue to bound operation under the proposed changes so that there is no safety margin reduction. [***].

Therefore, the proposed changes do not involve a significant reduction in 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: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Project Director: Ledyard B. Marsh

Sacramento Municipal Utility District (SMUD), Docket No. 50-312, Rancho Seco Nuclear Station, Sacramento County, California

Date of amendment request: June 20, 1995 and as amended August 14, 1995

Description of amendment request: The proposed amendment (PA-191) would permit SMUD to change the Fuel Storage Building load handling limits to allow placing the shield plugs on the dry shielded cannisters in order to permit transfer of spent fuel assemblies from the spent fuel pool (SFP) to the Rancho Seco Independent Spent Fuel Storage Installation.

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

PA-191 will not create a significant increase in the probability or consequences of an accident previously evaluated in the Safety Analysis Report (SAR), because dropping the dry shielded canister (DSC) top shield plug over a DSC loaded with 24 spent fuel assemblies is not considered a credible event. Also, the gantry crane is designed such that it can only handle loads over the SFP cask pit area and can not move a load over the SFP fuel storage racks.

PA-191 will not create the possibility of a new or different type of accident than previously evaluated in the SAR, because the proposed Permanently Defueled Technical Specification heavy load handling exceptions do not create a new credible accident scenario. Dropping the DSC top shield plug and damaging spent fuel assemblies is not considered a credible event.

PA-191 will not involve a significant reduction in the margin of safety, because the proposed heavy load handling exceptions do not create a credible accident scenario.

The NRC staff has reviewed the licensee's analyses of June 20, 1995 and August 14, 1995. The August 14 submittal enhanced these analyses by providing design details regarding the significant safety factors built into the crane and other lifting hardware. 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: Central Library, Government Documents 828 I Street, Sacramento, CA 95814

Attorney for licensee: Dana Appling, Esq. Sacramento Municipal Utility District, P. O. Box 15830, Sacramento, CA 95852-1830

NRC Project Director: Seymour H. Weiss

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 amendment requests: July 17, 1995

Description of amendment requests: The licensee proposes to revise surveillance requirements associated with Technical Specifications 3/4.3.1, ''Reactor Protective Instrumentation,'' and 3/4.3.2, ''Engineered Safety Feature Actuation System Instrumentation.'' The surveillance interval is to be increased to 120 days for performance of channel functional tests for certain reactor protective system and engineered safety feature actuation system instrumentation. The proposed change also revises Bases 3/4.3.1, ''Reactor Protective and Engineered Safety Features Actuation System Instrumentation,'' to reflect the new interval.

Basis for proposed no significant hazards consideration determination: As requied 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 would extend the current sequential Channel Functional Test (CFT) surveillance interval for Plant Protective System (PPS) instrumentation and Nuclear Instrumentation (NI). This change does not involve any changes to plant equipment or operation. The proposed change actually maintains or decreases the PPS system unavailability. PPS uncertainty and setpoint modifications will account for the new surveillance interval. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

This amendment request does not involve any change to plant equipment or operation. The PPS system is used for monitoring and mitigation of evaluated accidents. Increasing the availability of the PPS system, as proposed in this amendment request, will not create the possibility of a new or different kind of accident from any previously evaluated.

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

This amendment does not change the manner in which safety limits, limiting safety settings, or limiting conditions for operation are determined. This amendment request will increase Reactor Protective System and Engineered Safety Features Actuation System availability. Therefore, this amendment will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: August 7, 1995 (TS 95-12)

Description of amendment request: The proposed change would correct various errors of an editorial nature that have been identified in the technical specifications and remove the provisions that have exceeded their allowed time interval for implementation or the required conditions no longer exist.

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:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed revisions do not change the TS requirements, plant setpoints or functions, or plant operating practices. These changes provide clarifications to the existing TSs by correcting editorial errors and removing provisions that no longer apply in the specifications. The probability or consequences of an accident will not be increased by providing the proposed verbiage corrections that are editorial and nonintent.

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

No plant functions or compliance activities associated with the TS requirements have been affected by the proposed editorial changes. Therefore, the possibility of a new or different kind of accident is not created.

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

The proposed changes will not alter TS setpoint values or functions. The proposed corrections will enhance the application of TS requirements and will support the margin of safety provided by the TSs. Therefore, the margin of safety will not be reduced by the proposed revisions.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: August 7, 1995 (TS 95-17)

Description of amendment request: The proposed change would relocate the heat flux hot channel factor penalty of two percent from Surveillance Requirement 4.2.2.2.e.1 to the Core Operating Limits Report and add a reference to the factor to Specification 6.9.1.14.5. Also, Specification 6.9.1.14.a.2 would be revised to reference Revision 1A of Westinghouse Commercial Atomic Power (WCAP) 10216-P-A, ''Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specifications,'' dated February 1994.

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:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed change involves only the manner in which the penalty factors for $F_Q(Z)$ would be specified (i.e, a burnup-dependent factor specified in the Core Operating Limits Report [COLR] versus a constant factor specified in the TS). This is simply used to account for the fact that $F_Q(Z)$ may increase between surveillance intervals. These penalty factors are not assumed in any of the initiating events for the accident analyses. Therefore, the proposed change will have no effect on the probability of any accidents previously evaluated. The penalty factors specified in the COLR will be calculated using NRC-approved methodology and will therefore continue to provide an equivalent level of protection as the existing TS requirement. Therefore, the proposed change will not affect the consequences of any accident previously evaluated.

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

The proposed change does not involve a physical alteration to the plant (no new or different kind of equipment will be installed) or alter the manner in which the plant would be operated. 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 a margin of safety.

The proposed change will continue to ensure that potential increases in $F_Q(Z)$ over a surveillance interval will be properly accounted for. The penalty factors will be calculated using NRC-approved methodology. Therefore, the proposed change will not involve a reduction in margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: August 7, 1995 (TS 95-18)

Description of amendment request: The proposed change would revise the titles of various administrative positions found in Section 6.0 of the Technical Specifications.

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

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c).

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed changes only involve the administrative titles of management positions in TVA [Tennessee Valley Authority]. Plant equipment and operating practices are not affected by the proposed administrative changes. Therefore, there is no increase in the probability or consequences of an accident.

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

Plant features are not impacted by the proposed revision; therefore, this revision can not create the possibility of a new or different accident.

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

Plant setpoints and features that establish and maintain the margin of safety for SQN are not involved in the proposed administrative TS change. Therefore, the margin of safety is not reduced by the proposed change.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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 Romm location: Chattanooga-Hamilton County Library,1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: August 7, 1995 (TS 95-03)

Description of amendment request: The proposed change would modify Technical Specifications (TS) 3/4.1.3, ''Movable Control Assemblies,'' and Bases 3/4.1.3. The proposed change addresses operation with a rod urgent failure condition (the control rods are out-of-service because of failures external to the individual rod drive mechanisms; i.e., programming circuitry, but the control rods remain operable), including limited operation with one control or shutdown bank inserted up to 18 steps below its insertion point. In addition, the surveillance interval for rod movement verifications would be increased from 31 days to 92 days.

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:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c).

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

Allowing for continued operation during diagnosis and repair as a result of electronic or electrical malfunctions of the rod control system is acceptable, since the design safety function of the control rods (reactor trip will remain unaffected during the diagnosis and repair period. During the extended troubleshooting and repair period, the requirements for control rod alignment, insertion limits (except for a small allowed deviation for one bank) and shutdown margin will be maintained. The small deviation from the control rod insertion limits allowed for one bank, for up to 72 hours, will not adversely impact the current TS requirements for normal operation core power distributions. The proposed changes do not affect the ability of the control rods to perform their intended safety function (rods remain trippable) when a safety system setting is reached. No new or unique accident precursors be introduced by the proposed changes. Therefore, the probability and consequences of accidents related to or dependent on control rod operation will remain unaffected.

The proposed change will result in a small increase in the probability, that at any given time, a control or shutdown bank will be inserted slightly below (i.e., up to 18 steps) its insertion limit. However, by design, the control and shutdown banks will continue to meet the safety analysis criterion for steady state and American Nuclear Society (ANS) Condition II (moderate frequency) transients. The allowed insertion is not a malfunction of equipment important to safety in this case; therefore, the probability of such a malfunction is not increased. Limiting the allowed time for operation with the rod control system out-of-service, but with the rods trippable and with a control or shutdown bank below the insertion limit, eliminates the need for consideration of this condition coincident with any of the low frequency (ANS Condition III or IV) design basis accidents.

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

There are no new failure mechanisms associated with plant operation for an extended period to perform diagnosis and repair on the rod control system. Limited periods of operation with immovable, but trippable control rods, does not involve any modification to the operational limits or physical design of the involved systems. There are no new accident precursors created because of the allowed diagnosis and repair period.

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

The results of the current accident analyses are not impacted by the change. In addition, the margin of safety as defined in the basis of the TS has not been reduced because current core design limits continue to be met for the accidents of concern. Therefore, the margin of safety is not impacted.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: June 23, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirements 4.1.3.1.2, 4.4.6.2.2.b, 4.4.3.2, 4.6.2.1.d, 4.6.4.2, and Table 4.3-3 in accordance with guidance provided in NRC Generic Letter (GL) 93-05, ''Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operations.'' Additionally, the proposed amendment would revise TS 4.1.1.1.1, 4.1.1.2, 3/4.1.3.1 and associated Bases to implement portions of the Standard Technical Specifications - Westinghouse Plants, NUREG-1431.

Basis for proposed no significant hazards consideration determination: As requied 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 Technical Specification changes do not involve a significant hazards consideration per 10 CFR 50.92 because operation of Callaway Plant with the changes would not:

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

All changes are in accordance with the recommendations of NRC Generic Letter 93-05, Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation or NUREG 1431, Standard Technical Specifications - Westinghouse Plants. None of the changes affects accident initiators and each has been evaluated against Callaway Plant operating experience.

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

The proposed Technical Specification changes do not modify any equipment nor create any potential accident initiators. The changes per GL 93-05 involve Technical Specification surveillance frequencies and do not alter the methodology nor associated acceptance criteria. The changes per NUREG-1431 do not create any accident initiators and are consistent with Callaway design and operation.

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

The surveillance frequency changes were recommended via GL 93-05 and are compatible with Callaway Plant experience. The changes per NUREG-1431 do not impact the margin of safety. The Shutdown margin requirements and associated safety margins are unaffected by these changes.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 26, 1995

Description of amendment request: The proposed amendment would revise the allowed outage time for component cooling water motor operated containment isolation valves, remove the list of containment isolation valves, and allow containment penetration check valves to be used as isolation devices.

Basis for proposed no significant hazards consideration determination: As requied 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 revision to TS 3/4.6 to remove the listing of containment isolation valves, revise the ACTION Statement for the CCW MOVs, and credit penetration check valves as isolation devices does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

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

The proposed changes simplify the TS, meet the regulatory requirements for control of containment isolation and are consistent with the guidelines of GL 91-08. The information contained in Table 3.6-1 has not been changed, but only relocated to a different controlling document. This is an administrative change which should result in improved plant practices and have no impact on plant operations. Addition of the footnote to allow up to 12 hours for valve testing does not affect the severity of any accident previously evaluated. The proposed revision to the TS will not adversely impact plant safety since the second barrier of the two required is still available to provide isolation between the containment atmosphere or the reactor coolant system and the outside atmosphere.

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

There are no design changes being made that would create a new type of accident or malfunction and the method and manner of plant operation remain unchanged. Addition of the footnote to allow up to 12 hours for valve testing does not affect the severity of any accident previously evaluated. The additional time provides assurance that the inoperable valve is in proper working order prior to returning it to OPERABLE condition.

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

There are no changes being made to the safety limits or safety system settings that would adversely impact plant safety. Containment isolation will still be maintained as provided by the second isolation valve to ensure that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. This will assure that containment integrity is maintained.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: July 25, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.8.1 and its associated Bases to improve overall emergency diesel generator reliability and availibility.

Basis for proposed no significant hazards consideration determination: As requied 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 changes do not involve a significant hazards consideration because operation of Callaway Plant with these changes would not:

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

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. Emergency diesel generator operability and reliability will continue to be assured while minimizing the number of required emergency diesel generator starts. Also, emergency diesel generator reliability will be enhanced by minimizing severe test conditions which can lead to premature failures.

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

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. There is no actual impact on any accident anaiysis.

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

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. No margin of safety is 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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: November 29, 1994

Description of amendment request: The proposed change would revise and update the NA-1&2 Environmental Protection Plan (EPP) to reflect current obligations to the Commonwealth of Virginia, revise portions of the transmission corridor rights-of-way erosion control program for clarification and to be consistent with the state regulations, eliminate inconsistencies, and delete obsolete material. Specifically, references to National Pollutant Discharge Elimination System (NPDES) permits are changed to reflect the correct permit title, Virginia Pollutant Discharge Elimination System (VPDES). Vegetation and aquatic biota studies referred to in the EPP were satisfactorily completed on or before June 24, 1986. The discussion of the detailed subject matter in these studies is removed because it is extraneous information. A reference to 10 CFR 51.5(b)(2) (which does not exist) is corrected to 10 CFR 51.60(b)(2). The explicit reporting requirements for unusual or important environmental events are replaced with the reporting requirement which the NRC has required pursuant to 10 CFR 50.72 (b)(2)(vi). Therefore, the reporting inconsistency (EPP requires report to NRC within 24 hours, whereas the 10 CFR 50.72 requires a four hour report to the NRC) is resolved. The description of the audit program to be utilized for auditing the EPP is replaced by referring to the Audit Program established in accordance with 10 CFR 50, Appendix B. Another inconsistency is eliminated by revising the two year records retention requirement for erosion control inspection field logs to five years. This makes the requirement consistent with EPP Section 5.2, Records Retention. References to the State Water Control Board are updated to that agency's successor, the Department of Environmental Quality. Additionally, the licensee's obligation to comply with Virginia regulations concerning erosion and sediment control within the transmission corridor rights-of-way are recognized to eliminate redundancy with previous EPP commitments. The Virginia Soil and Water Conservation Board is recognized as the regulatory authority concerning erosion within the transmission corridor rights-of-way. The Virginia Soil and Water Conservation Board reviews and approves erosion and sediment control specifications submitted by utilities on an annual basis.

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

Specifically, operation of the North Anna Power Station in accordance with the EPP changes will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The likelihood that an accident will occur is neither increased or decreased by the proposed changes to the EPP. Sufficient controls are established to ensure that environmental controls impacting safety-related structures, systems, and components are maintained current and accurate. The only potentially credible accident which might be affected is the Loss of Offsite Power (if erosion were severe

enough to undermine the bases of a transmission tower). Each of the three 500 KV transmission lines connected to North Anna Power Station can supply sufficient power to the site. This limits the effect that one transmission tower has on safe operation of the nuclear facility. However, the erosion noted to date has not been severe enough to make such an accident credible. Additionally, each of the 500 KV transmission lines are inspected for material condition annually. Although the intent of this inspection is not soil erosion (the annual erosion inspections are currently conducted by another group who specializes in land management), evidence of severe erosion would be noted and addressed as appropriate. Therefore, this EPP change will not impact the function or method of operation of plant equipment. Thus, a significant increase in the probability of a previously analyzed accident does not result due to this change. Nuclear station systems, equipment, or components are not affected by the proposed changes. Thus, the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are not increased by this change.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve changes to the physical plant or operations. ... the proposed EPP changes do not contribute to accident initiation and therefore do not produce a new accident scenario or produce a new type of equipment malfunction. Also, this EPP change does not alter any existing accident scenarios. The proposed changes do not affect nuclear plant equipment or its operation, and thus do not create the possibility of a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

(3) Involve a significant reduction in a margin of safety. The EPP does not have a formal basis description other than the discussion in the FES-OL [Final Environmental Statement-Operating License]. The FES-OL discusses the non-radiological impacts of facility construction and operation on the environment. The discussion indicates that the environment will be managed to a stabilized condition during the operations phase, and a program will be implemented to maintain the environment in a stabilized condition. This intent is not altered by the proposed changes to the EPP. The proposed changes do not affect nuclear plant equipment or its operation, and thus do not involve any reduction in the margin of safety.

Therefore, use of the proposed EPP would not involve any reduction in the margin of safety.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498

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

NRC Project Director: David B. Matthews

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

Date of amendment request: July 26, 1995

Description of amendment request: The proposed changes would revise the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the proposed changes would increase the pressurizer safety valve lift setpoint tolerance as well as reduce the pressurizer high pressure reactor trip setpoint and allowable value.

The licensee has prepared a safety evaluation which justifies increasing the current TS pressurizer safety valve (PSV) at-power (Modes 1-3) lift setpoint tolerance from plus or minus 1% as-found and plus or minus 1% as-left to +2%/-3% average as-found with no single valve outside plus or minus 3% as-found and plus or minus 1% per valve as-left. The as-found value is based on testing, the results of which are expressed as an error (i.e., positive or negative percentage deviation from the nominal lift setpoint). The errors of the tested valves are summed and the result divided by the number of valves tested. This result is compared to the acceptable range of +2% to -3%. No single valve is allowed to be outside of the plus or minus 3% tolerance.

The safety evaluation also supports an increase to the Hot Shutdown (Mode-4) required PSV lift setpoint tolerance from plus or minus 1% as-found and plus or minus 1% as-left to plus or minus 3% per valve as-found and plus or minus 1% per valve as-left. These proposed changes will provide greater operational flexibility in meeting periodic test requirements established by the safety analyses.

A concurrent reduction in the pressurizer high pressure reactor trip setpoint and allowable value of TS Table 2.2-1 are also proposed. These changes ensure that the analysis results for the loss of external load accident continue to meet the acceptance criteria with the higher PSV tolerance.

The Loss of Load, Locked Rotor, and Rod Withdrawal event analyses demonstrate that increasing the at-power PSV lift setpoint tolerance to +2%/-3% average as-found with no single valve outside plus or minus 3% as-found and plus or minus 1% per valve as-left does not result in a transient pressure in excess of the overpressure safety limit. Further, the increased setpoint tolerance does not adversely impact the DNBR [departure from nucleate boiling ratio] results of any North Anna UFSAR [Updated Final Safety Analysis Report] Chapter 15 transient analysis. Mode 4 overpressure protection is adequate with one PSV with a tolerance of plus or minus 3%.

Finally, the increased PSV setpoint tolerances and reduction of the high pressurizer pressure reactor trip setpoint do not present any operational considerations which would significantly impact the performance of the plant during normal operation or during postulated accident conditions. In summary, each pertinent safety criterion was evaluated for the proposed TS changes, and all were found to be acceptable.

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

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

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

Affected safety related parameters were analyzed for a change to North Anna 1 and 2 Technical Specifications 3.4.2 and 3.4.3 and Table 2.2-1 item 10. It was determined that the overpressure safety limits would not be exceeded in the most limiting overpressure transients (Loss of Load, Locked Rotor, and Rod Withdrawal events) with the as-found pressurizer safety valve lift setpoint tolerance increased to an average of +2%/-3%, no single valve outside of [plus or minus] 3%, and the 25 psi reduction in the Pressurizer High Pressure Reactor Trip setpoint. The DNBR results of transients impacted by the proposed setpoint tolerance increase meet the acceptance criterion after accounting for the impact of the proposed changes. The increased setpoint tolerance will not result in an inadvertent opening of the pressurizer safety valves. Mode 4 overpressure protection is adequate with one PSV with a tolerance of [plus or minus] 3%.

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

The proposed change to North Anna 1 and 2 Technical Specifications 3.4.2 and 3.4.3 and Table 2.2-1 item 10 does not involve any changes which would introduce any new or unique operational modes or accident precursors. Only the allowable tolerance about the existing PSV lift setpoint will be changed, along with a reduction in the pressurizer high pressure reactor trip setpoint.

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

It was determined that the most limiting overpressure transients do not result in maximum pressures in excess of the overpressure safety limits. The DNBR results of transients impacted by the proposed setpoint tolerance increase meet the acceptance criterion after accounting for the impact of the proposed changes. Therefore, the margin of safety is unchanged by the proposed increase in the safety valve setpoint tolerances.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498

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

NRC Project Director: David B. Matthews

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

Date of amendment request: July 26, 1995

Description of amendment request: The proposed change would revise the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the change would clarify the TS to allow switching of charging and low-head safety injection pumps during unit shutdown conditions. The proposed changes would also allow additional methods of rendering these same pumps incapable of injecting into the reactor coolant system (RCS) when required for low-temperature conditions. NA-1&2 is equipped with three charging pumps. These charging pumps provide inventory control, normal boration to the RCS, and flow to the reactor coolant pump seals. They also act as the high-head safety injection pumps during accident conditions. During certain shutdown conditions, it is necessary to render two of the three charging pumps inoperable to maintain the low-temperature overpressure protection (LTOP) design bases assumptions. This provides assurance that a mass addition pressure transient can be relieved by the operation of a single pressurizer power-operated relief valve (PORV). Low-temperature overpressure protection for each NA-1&2 unit is provided by two pressurizer PORVs.

During shutdown conditions, periodic surveillance testing of the charging pumps is required by the NA-1&2 TS. Also during shutdown conditions, it may be desirable to switch from one charging pump to another to allow for other activities such as maintenance or testing.

The current NA-1&2 TS associated with charging pumps during shutdown conditions are very restrictive and do not allow sufficient latitude for surveillance testing or pump switching. The current NA-1&2 TS specifically state in the surveillance requirements that the method used to render a charging pump inoperable is to place the pump control switch in the pull-to-lock position. This requirement would not allow for surveillance or post-maintenance testing of the inoperable charging pumps since this switch is used to start those pumps.

Therefore, the licensee proposes to modify NA-1&2 TS to allow more than one charging pump to be operable and capable of injecting into the RCS for pump switching operations. Additionally, the methods used to render charging pumps inoperable will be expanded to allow for post-maintenance and surveillance testing.

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

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

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

Allowing more than one charging pump to be operable and capable of injecting into the RCS during RCS low temperature operation for pump switching for post-maintenance and surveillance testing does not increase the probability of occurrence or the consequences of any previously analyzed accident. Pump switching operations will be under the direct administrative control of a licensed operator and will only be for a short duration of time. Any situation that could result in an excessive RCS mass addition would be immediately recognized by the operator and remedial action would be taken to prevent challenges to RCS integrity. Using methods such as opening the charging pump power supply breaker or closing the charging pump discharge valve(s) to render a charging pump inoperable will ensure that these pumps will not be capable of injecting water into the RCS. These alternate methods are as effective as placing the control switches in the pull-to-lock position.

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

Allowing more than one charging pump to be operable and capable of injecting into the RCS during low-temperature operation for pump switching for post-maintenance and surveillance testing does not involve any physical modifications of the plant nor result in a change in a method of operation. Licensed operator control of charging pump switching operations will continue to ensure that the RCS will not be challenged by excessive mass addition events. Using methods other than placing charging pump control switches in the pull-to-lock position to render the pump inoperable will still ensure that only one pump will be capable of injecting into the RCS during low temperature operations. Therefore, a new or different type of accident is not made possible.

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

Allowing more than one charging pump to be operable and capable of injecting into the RCS during RCS low temperature operation for pump switching for post-maintenance and surveillance testing does not affect any safety limits or limiting safety system settings. The alternate methods of rendering pumps inoperable provide the same level of assurance that the pump is incapable of flowing into the RCS as placing the pump control switch in the pull-to-lock position. System operating parameters remain unaffected. The availability of equipment required to mitigate or assess the consequence of an accident is not reduced. Safety margins are, therefore, not decreased.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498

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

NRC Project Director: David B. Matthews

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

Date of amendment request: July 20, 1995

Description of amendment request: The proposed amendments would: 1) revise three Reactor Protection System/Engineered Safety Features Actuation Systems channel trip setpoint limits, 2) add a new setpoint limit for high high steam generator water level, and 3) incorporate editorial changes to revise the measurement units of one setpoint limit and to delete certain references to two-loop operation.

Basis for proposed no significant hazards consideration determination: As requied by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:Specifically, operation of Surry Power Station with the proposed change will not:

(1) Involve a significant increase in either the probability of occurrence or consequences of any accident or equipment malfunction scenario which is important to safety and which has been previously evaluated in the Updated Safety Analysis Report (UFSAR). The effect of the proposed change is to ensure that actual plant setpoints remain conservative consistent with respect to accident analysis assumptions. The proposed change requires safety system actuation limits that are more conservative than those currently in Technical Specifications. The change does not invalidate currently implemented station setpoints or currently applicable accident analysis assumptions regarding these setpoints. Consequently, the results and conclusions of the current UFSAR accident analyses are not affected by these changes. The proposed Technical Specifications change revises setpoints used to mitigate accidents and therefore has no bearing on the probability of an accident. Further, the change ensures that the setpoints used to mitigate an accident bound the setpoints used in the accident analyses. Therefore, the probability of an accident or consequences of an accident is not adversely affected as a result of this change.

(2) Create the possibility of a new or different type of accident than those previously evaluated in the UFSAR. Implementing the proposed Technical Specifications setpoint limits cannot create the possibility of an accident of a different type than was previously evaluated in the UFSAR. Since actual plant setpoints are not being affected, new accident precursors will not be introduced. Furthermore, spurious challenges to safety systems are also not expected to increase in frequency as a result of these changes since actual setpoints installed in the plant are not being changed. Consequently, no new accident precursors are created as a result of the new Technical Specifications setpoint limits.

(3) Involve a significant reduction in a margin of safety. Since the results of the existing UFSAR accident analyses remain bounding, safety margins are not impacted. The proposed Technical Specifications setpoint limits ensure plant setpoints remain conservative and consistent with design base accident analysis assumptions including appropriate instrument channel uncertainties due to harsh environmental conditions. Therefore, the margin of safety as defined in the Technical Specifications bases is unaffected.

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: Swem Library, College of William and Mary, Williamsburg, Virginia 23185

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

NRC Project Director: David B. Matthews

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

Date of amendment request: July 25, 1995

Description of amendment request: This license amendment request proposes to revise Technical Specification 4.0.5a and Bases Section 3/4.4.10 to delete the clause ''(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' This proposed change is consistent with NUREG-1482, ''Guidelines for Inservice Testing and Nuclear Power Plants.''

Basis for proposed no significant hazards consideration determination: As requied 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.

This proposed change would remove the wording ''...(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed change, in accordance with NUREG-1431 and NUREG-1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted the relief. The change being proposed is administrative in nature and does not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor does it affect any technical specification that preserves safety analysis assumptions. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed change would not affect the probability or consequences of an accident previously analyzed.

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

This proposed change would remove the wording ''...(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed change, in accordance with NUREG-1431 and NUREG-1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC had issued a safety evaluation and granted the relief. The change being proposed is administrative in nature and will not change the physical plant or the modes of operation defined in the facility license. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change would remove the wording ''...(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).'' The Inservice Inspection and Testing Programs are described in the technical specifications pursuant to 10 CFR 50.55a. In addition, the proposed change, in accordance with NUREG-1431 and NUREG-1482, would provide relief to the ASME Code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted the relief. The change being proposed is administrative in nature and will not alter the bases for assurance that safety-related activities are performed correctly or the basis for any technical specification that is related to the establishment or maintenance of a safety margin. Any relief from the approved ASME Section XI Code requirements will require a 10 CFR 50.59 evaluation to ensure no technical specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed change 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: William H. Bateman

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

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

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

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 11, 1995

Description of amendment request: The proposed amendment would remove Technical Specification Section 3.2, ''Makeup and Purification and Chemical Addition Systems,'' and its bases. The pertinent requirements and bases applicable to these systems are being incorporated in the TMI-1 Updated Final Safety Analysis Report (UFSAR).

Date of publication of individual notice in **Federal Register:** August 18, 1995 (60 FR 43172)

Expiration date of individual notice: September 18, 1995

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

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: April 6, 1995, and superseded on August 7, 1995

Description of amendments request: Amend the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specification (TS) to revise the numerical values for the overtemperature and overpower delta-temperature equation constants in TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints.

Date of publication of individual notice in the **Federal Register:** August 15, 1995 (60 FR 42187)

Expiration date of individual notice: September 14, 1995

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: May 2, 1995

Brief description of amendments: The amendments remove from the technical specifications (TS) plant elevations for the minimum water volume required in the spent fuel pool and relocate them to site procedures. The TS amendment also includes two changes to correct administrative errors in the TS.

Date of issuance: August 7, 1995

Effective date: August 7, 1995

Amendment Nos.: Unit 1 - Amendment No. 97 ; Unit 2 - Amendment No. 85; Unit 3 - Amendment No. 68

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

Date of initial notice in **Federal Register:** July 5, 1995 (60 FR 35060) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 7, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: January 25, 1993, as supplemented on December 28, 1993, September 13, 1994, January 13, 1995, and May 25, 1995. The supplemental submittals did not expand the scope of the original **Federal Register** notice or change the no significant hazards determination.

Brief description of amendments: The amendments allow unit entry into Operational Condition 1 (Power Operation) from Operational Condition 2 (Startup) with up to eight inoperable control rods, provided those control rods are not inoperable due to being immovable or untrippable.

Date of issuance: August 11, 1992

Effectove date: August 11, 1992

Amendment Nos.: 178 and 209

Facility Operating License Nos. DPR-71 and DPR-62.

Date of initial notice in **Federal Register:** July 7, 1993 (58 FR 36428) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 11, 1995.Significant hazards consideration comments received: No.

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

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: April 5, 1995, as supplemented July 31, 1995

Brief description of amendment: The amendment revises various portions of TS 3/4.9, Refueling Operations, to be consistent with NUREG-1431, ''Standard Technical Specifications, Westinghouse Plants,'' and allows the relocation of applicable sections from the TS that do not meet the Commission screening criteria for retention.

Date of issuance: August 9, 1995

Effectove date: August 9, 1995

Amendment No.: 61

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

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24906) The July 31, 1995 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

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

Date of application for amendments: January 13, 1995

Brief description of amendments: The amendments revise the pressure alarm setpoint allowable values for the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system ''keep filled'' pressure instrumentation channels. The purpose of the change is to lower the setpoint allowable values for these parameters to more realistic values based upon calculations performed by the licensee reflecting design changes and system performance. Also, the term ''setpoint'' is being changed to ''setpoint allowable value'' to clarify the use of the values. Additionally, two administrative/editorial changes are included to delete technical specification footnotes which are no longer applicable.

Date of issuance: August 15, 1995

Effectove date: Immediately, to be implemented within 90 days.

Amendment Nos.: 105 and 91

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11128) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1995.No significant hazards consideration comments received: No

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

Duke Power Company, 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: February 24, 1994, as supplemented by letters dated April 19, May 25, August 25, 1994, January 4, January 27, February 22, March 15, April 19, and May 31, 1995

Brief description of amendments: The amendments provide surveillance requirements for a planned modification to the Keowee emergency power generators' underground power path breaker closing logic.

Date of issuance: August 15, 1995

Date of issuance: August 15, 1995

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

Amendment Nos.: 210, 210, and 207

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

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14887) The April 19, May 25, August 25, 1994, January 4, January 27, February 22, March 15, April 19, and May 31, 1995, letters provided clarifying information that did not change the scope of the February 24, 1994, application and initial no proposed significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 30, 1995, as supplemented May 5, 1995 and June 19, 1995

Brief description of amendments: These amendments relate to separation of the 24-hour emergency diesel generator test and hot restart test from the loss of offsite power test.

Date of issuance: August 8, 1995

Effectove date: August 8, 1995

Amendment Nos.: 175 and 169Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 23, 1995 (60 FR 27339), and July 5, 1995 (60 FR 35072) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 8, 1995.No significant hazards consideration comments received: No

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

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, 1995, as supplemented by letters on May 20, 1994, and March 8, 1995

Brief description of amendment: The amendment revises Technical Specification Section 6.5.3, ''AUDITS,'' by removing the specified frequency for internal audits. These frequency specifications will now be located in Appendix E of the GPU Nuclear Operational Quality Assurance Plan (1000-PLN-7200.01). A minor editorial change has been incorporated into TS 6.5.1.14 correcting a reference in response to a finding in the Operational Safety Team Inspection (OSTI) report of December 23, 1993.

Date of issuance: August 7, 1995

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

Amendment No.: 181

Facility Operating License No. DPR-16. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 25, 1994 (59 FR 27056)

The letters of May 20, 1994, and March 8, 1995, provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 7, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 19, 1994, supplemented March 8, 1995

Brief description of amendment: The amendment revises the TMI-1 Technical Specification (TS) Section 6.5.3 to remove the specified frequency of various licensee-conducted audits, including those related to quality assurance, fire protection, security, emergency preparedness, and offsite dose calculations. The frequencies for conduct of these audits will now be specified in the licensee's Operational Quality Assurance Plan, which requires NRC approval for significant changes. The Commission has determined that these audit frequencies need not be in the TS to assure public health and safety.

Date of issuance: August 14, 1995

Effectove date: August 14, 1995

Amendment No.: 195

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29627) The March 8, 1995, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 14, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 13, 1994, as supplemented April 3, 1995

Brief description of amendment: The amendment revises Table 3.6.1.2-1 to allow a maximum leakage of 24.0 scfh for each of the 8 main steam isolation valves instead of the current 6.0 scfh.

Date of issuance: August 10, 1995

Effectove date: As of the date of issuance to be implemented within 60 days

Amendment No.: 67

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

Date of initial notice in **Federal Register:** January 18, 1995 (60 FR 3675) The April 3, 1995, letter provided clarifying information that did not change the initial no proposed significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 10, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 15, 1995 (published in Federal Register as March 15, 1994) as supplemented by letter dated August 5, 1995

Brief description of amendments: These amendments modify the Susquehanna Steam Electric Station Technical Specification Table 3.6.3-1, Primary Containment Isolation Valves, concerning the scope of Type C testing on specified emergency core cooling system and reactor core isolation cooling containment isolation valves. Specifically, the subject valves on systems which terminate below the minimum water level of the suppression pool will no longer require Type C testing but will instead be tested using requirements of the American Society of Mechanical Engineers' Section XI Code.

Date of issuance: August 15, 1995

Effectove date: August 15, 1995

Amendment Nos.: 149 and 119

Facility Operating License Nos. NPF-14 and NPF-22. The amendments revised the Technical Specifications.The supplemental letter did not change the proposed no significant hazards consideration determination nor the Federal Register notice.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20521) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 31, 1995, as supplemented by letter dated June 22, 1995

Brief description of amendments: These amendments delete from the Technical Specifications of each unit, the operational condition restriction in Surveillance Requirement 4.8.1.1.2.d.7, which requires that 24-hour emergency diesel generator testing be performed with at least one unit in operational condition 4 or 5 (cold shutdown or refueling).

Date of issuance: August 15, 1995

Effectove date: Units 1 and 2, effective as of the date of issuance and shall be implemented within 60 days

Amendment Nos.: 150 and 120

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

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20523) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 21, 1994, as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995

Brief description of amendments: These amendments change the Technical Specifications for the two units by deleting reference to the main steamline isolation valve (MSIV) leakage control system and its associated primary containment isolation valves, and increase the allowable leakage rate for any MSIV and the total maximum pathway leakage for all four main steam lines.

Date of issuance: August 15, 1995

Effectove date: Units 1 and 2 as of date of issuance and shall be implemented within 30 days

Amendment Nos.: 151 and 121

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

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 503) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 2, 1995

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

Date of issuance: August 8, 1995

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

Amendment No.: 226

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

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24916) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: February 5, 1993, supplemented April 13, June 11 and November 17, 1993

Brief description of amendments: The amendment eliminates the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level Reactor Trip due to the installation of the digital feedwater control system incorporating a median signal selector.

Date of issuance: August 7, 1995

Effectove date: Unit 1, as of the date of issuance, to be implemented by the startup following the twelfth refueling outage, Unit 2, as of the date of issuance, to be implemented by the startup following the current outage

Amendment Nos.: 173 and 154

Facility Operating License Nos. DPR-70 and DPR-75: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 28, 1993 (58 FR 25864) The April 13, June 11, and November 17, 1993 submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1995.No significant hazards consideration comments received: No

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

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

Date of amendment request: March 11, 1994

Description of amendment request: The amendment decreases the allowable time for operation with one inoperable residual heat removal (RHR) relief valve from 7 days to 72 hours. This amendment request has been submitted in response to Generic Issue 94 as discussed in Generic Letter 90-06.

Date of issuance: August 11, 1995

Effectove date: August 11, 1995

Amendment No.: 125

Facility Operating License No. NPF-12: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** June 22, 1994 (59 FR 32236) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, South Carolina 29180

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

Date of amendment request: February 14, 1994 (TXX-94045), as supplemented by letter dated May 23, 1995 (TXX-95147)

Brief description of amendments: The amendments incorporated appropriate references to and provisions of the new 10 CFR Part 20 regulations. These changes revised a definition and aspects of radiological effluent technical specifications, clarified the administrative specification for reporting individual annual exposures greater than 100 mrem by work/job function, and revised the administrative specifications for providing alternative measures for control of access to high radiation areas and designating record retention for radioactive shipments.

Date of issuance: August 11, 1995

Effectove date: August 11, 1995

Amendment Nos.: Unit 1 - Amendment No. 42; Unit 2 - Amendment No. 28

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

Date of initial notice in **Federal Register:** April 28, 1994 (59 FR 22016) The additional information contained in the supplemental letter dated May 23, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determinations. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 11, 1995.No significant hazards consideration comments received: No.

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

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: March 31, 1994, as supplemented by letters dated September 9, 1994, and June 22, 1995

Brief description of amendment: The amendment modifies the requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with the Boiling Water Reactor (BWR) Owners Group long-term solution Option I-D described in the Licensing Topical Report, ''BWR Owners Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960 June 1991'' and NEDO-31960, Supplement 1, Dated March 1992. NEDO-31960 and NEDO-31960, Supplement 1, were accepted by the NRC staff in a letter to L.A. England (BWR Owners Group) dated July 12, 1993.

Date of issuance: August 9, 1995

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

Amendment No.: 146

Facility Operating License No. DPR-28. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 507) The September 9, 1994, and June 22, 1995, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

Dated at Rockville, Maryland, this 23rd day of August.

For The Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 95–21389 Filed 8–29–95; 8:45 am]

BILLING CODE 7590–01–F

[Docket No. 40–0299]

Federal Register Notice of Amendment To Change Reclamation Milestone Dates in Source Material License SUA–648 Held by Umetco Minerals Corporation

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Amendment of Source Material License SUA–648 to change reclamation milestone dates.

SUMMARY: Notice is hereby given that the U.S. Nuclear Regulatory Commission has amended Umetco Mineral Corporation's (Umetco's) Source Material License SUA–648 to change the reclamation milestone dates. This amendment was requested by Umetco by letter dated April 21, 1995, and its receipt by NRC was noticed in the **Federal Register** on June 21, 1995.

The license amendment modifies License Condition 59 to change the completion dates for four site-reclamation milestones. The new dates approved by the NRC extend completion of (1) placement of final radon barrier on the A–9 Impoundment by one year, and (2) placement of erosion protection on the Inactive Impoundment, the A–9 Impoundment, and the Heap Leach Impoundment by one year. Umetco attributes the delays to (1) NRC's re-examination of cover design for performance with current standards and practices, and (2) short construction season at the Gas Hills site. Based on review of Umetco's submittal, the NRC staff concludes that the delays are attributable to factors beyond the control of Umetco, the proposed work is scheduled to be completed as expeditiously as practicable, and the added risk to the public health and safety is not significant.

An environmental assessment is not required since this action is categorically excluded under 10 CFR 51.22(c)(11), and an environmental report from the licensee is not required by 10 CFR 51.60(b)(2).

SUPPLEMENTARY INFORMATION: Umetco's license, including an amended License Condition 59, and the NRC staff's technical evaluation of the amendment request are being made available for public inspection at the Commission's Public Document Room at 2120 L Street, NW (Lower Level), Washington, DC 20555.

FOR FURTHER INFORMATION CONTACT: Mohammad W. Haque, High-Level Waste and Uranium Recovery Projects Branch, Division of Waste Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415–6640.

Dated at Rockville, Maryland, this 21st day of August 1995.

Joseph J. Holonich,

Chief, High-Level Waste and Uranium Recovery Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 95–21494 Filed 8–29–95; 8:45 am]

BILLING CODE 7590–01–M

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–36139; File No. SR–CHX–95–19]

Self-Regulatory Organizations; Notice of Filing of Proposed Rule Change and Amendment No. 1 to the Proposed Rule Change, by the Chicago Stock Exchange, Inc. Relating to the Chicago Match

August 23, 1995.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act''), 15 U.S.C. 78s(b)(1), notice is hereby given that on July 27, 1995, the Chicago Stock Exchange, Inc. (''CHX'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'' or ''SEC'') the proposed rule change, and on August 22, 1995, filed Amendment No. 1 to the proposed rule change,[1] as described in Items I, II and III below, which Items have been prepared by the self-regulatory organization. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The CHX proposes to amend Article XXVII of the Exchange's Rules to increase the number of daily matches in the Chicago Match to two.

II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the self-regulatory organization included statements concerning the purpose of and basis for the proposed rule change and discussed any comments it received on the proposed rule change. The text of these statements may be examined at the places specified in Item IV below. The self-regulatory organization has prepared summaries, set forth in Sections A, B, and C below, of the most significant aspects of such statements.

A. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

1. Purpose

On November 30, 1994, the Commission approved a proposed rule of the Exchange that created the Chicago Match, an institutional trading system

[1] *See* Letter from David T. Rusoff, Attorney, Foley & Lardner, to Elisa Metzger, Attorney, SEC, dated August 22, 1995.