application of the standard would result in a diminution of safety to the miners. In addition, the petitioner asserts that the proposed alternative method would provide at least the same measure of protection as would the mandatory standard.

9. Performance Coal Company

[Docket No. M–95–100–C]

Performance Coal Company, P.O. Box 69, Naoma, West Virginia 25140 has filed a petition to modify the application of 30 CFR 75.350 to its Upper Big Branch South Mine (I.D. No. 46–08436) located in Raleigh County, West Virginia. The petitioner proposes to install a low-level carbon monoxide detection system as an early warning fire detection system in all belt entries used as intake air courses. The petitioner asserts that the proposed alternative method would provide at least the same measure of protection as would the mandatory standard.

10. Performance Coal Company

[Docket No. M–95–101–C]

Performance Coal Company, P.O. Box 69, Naoma, West Virginia 25140 has filed a petition to modify the application of 30 CFR 75.1700 to its Upper Big Branch South Mine (I.D. No. 46–08436) located in Raleigh County, West Virginia. The petitioner proposes to plug and mine through oil and gas wells. The petitioner asserts that the proposed alternative method would provide at least the same measure of protection as would the mandatory standard.

11. C.L.D., Inc.

[Docket No. M–95–10–M]

C.L.D., Inc., 2765 East 500 South, Vernal, Utah 84078 has filed a petition to modify the application of 30 CFR 57.4760(a) to its Cowboy No. 1 and 2 Mine (I.D. No. 42–02096) located in Uintah County, Utah. The petitioner requests a variance from the mandatory safety standard because the mining methods used at its gilsonite mines do not provide a physical means to comply with the standard. The petitioner states that a gilsonite mine uses an open-trench method of mining and that the mines are connected to other mines and are self-ventilating. The petitioner asserts that the proposed alternative method would provide at least the same measure of protection as would the mandatory standard.

Request for Comments

Persons interested in these petitions may furnish written comments. These comments must be filed with the Office of Standards, Regulations and Variances, Mine Safety and Health Administration, Room 627, 4015 Wilson Boulevard, Arlington, Virginia 22203. All comments must be postmarked or received in that office on or before September 1, 1995. Copies of these petitions are available for inspection at that address.

Dated: July 25, 1995.

Patricia W. Silvey,

Director, Office of Standards, Regulations and Variances.

[FR Doc. 95–18947 Filed 8–1–95; 8:45 am]

BILLING CODE 4510–43–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 7, 1995, through July 21, 1995. The last biweekly notice was published on Wednesday, July 19, 1996 (60 FR 37084).

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 1, 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: July 3, 1995

Description of amendments request: The proposed Technical Specification (TS) amendment temporarily adds new ACTION Statements 3.8.1.1.f and 3.8.1.1.g to TS 3.8.1.1, ''A.C. Sources - Operating,'' to provide a method of responding to sustained degraded switchyard voltage. Bases 3/4.8.1, ''A.C. Sources,'' 3/4.8.2, ''D.C. Sources,'' and 3/4.8.3, ''Onsite Distribution Systems,'' are also being revised to provide guidance on how and why degraded offsite power voltage and the number of startup transformers in service affect compliance with GDC 17 and to give the basis for the additional ACTION statements.

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

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

The proposed change does not significantly increase the probability of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The safety function of the Electrical Distribution System (EDS) is to provide sufficient capacity and capability to assure that 1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and 2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. In addition, it shall have sufficient independence, redundancy, and testability to perform its safety function assuming a single failure. The proposed ACTIONs will restore the EDS to conformance with General Design Criterion (GDC) 17 of Appendix A to 10 CFR 50. Once in conformance with GDC 17, the system will be capable of performing its safety function as analyzed in Chapters 6 and 15 of the UFSAR. The proposed temporary change has no effect on the probability of accident initiation, therefore, the probability of an accident previously evaluated has not been significantly increased.

The consequences of an accident previously evaluated in the UFSAR will not be significantly increased. Restoring one train to OPERABLE, by blocking Fast Bus Transfer (FBT), within one hour is consistent with the response time of Technical Specification (TS) ACTION 3.0.3. The second train will be restored to OPERABLE by having its Emergency Diesel Generator (EDG) started, loaded, and separated from offsite power within two hours or FBT will be blocked within two hours. Action within two hours is consistent with the plant—s TS since TS ACTION 3.8.2.1.a, ''D. C. Sources - Operating,'' would be the most limiting requirement with one train of inoperable electric power. In a degraded voltage event, the ability of the Class 1E 125VDC battery chargers to perform their function is indeterminate, therefore, the Class 1E 125VDC batteries must be assumed to provide the 125VDC control power to the Class 1E Engineered Safety Features (ESF) circuit breakers for both of their sequences. The battery capacity calculations assume only one sequence. Once one train is restored to OPERABLE and the other train—s EDG demonstrated to be OPERABLE by loading and separating from the grid, ACTION 3.8.1.1.a, for one INOPERABLE offsite power supply, allows operation to continue for up to seventy-two hours. If both trains are blocked, then both trains are OPERABLE.

The proposed change will ensure that the train that blocks FBT will be in conformance with GDC 17 should a subsequent accident occur. As such, that train of ESF equipment will be supplied Class 1E preferred and standby power in the manner assumed by Chapters 6 and 15 analyses. Starting, loading, and separating the other train—s EDG from offsite power ensures that the second train of ESF equipment is prepared to respond to any subsequent accident. This configuration presents one OPERABLE offsite circuit and two OPERABLE EDGs to any subsequent accident, and would be capable of withstanding the single failures in the UFSAR Table 15.0-0, ''Single Failures.'' Optionally, with both trains blocked, both are OPERABLE and would be capable of withstanding the single failures in the UFSAR Table 15.0-0, ''Single Failures.''

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

Given the current licensing basis, the proposed temporary TS change does not create the possibility of an accident of a new or different kind. The plant is currently licensed to have both trains of FBT blocked when low switchyard voltages exist in order to prevent the loss of power generated by the nuclear power unit from causing the loss of the preferred power circuits. The proposed temporary TS ACTIONs 3.8.1.1.f and 3.8.1.1.g are being added as ACTIONs to prevent a double sequencing event from occurring. The train that is blocked is consistent with previous UFSAR Chapter 6 and Chapter 15 safety analyses since it will conform to GDC 17 prior to the onset of the accident. Under this condition it will be able to contribute to the mitigation of an accident and withstand the effects of any single failure equal to its ability when initially analyzed and licensed. The EDG which is loaded and isolated from offsite power also contributes to GDC 17 compliance since the entire system can withstand a Loss of Offsite Power (LOP) and a single failure of an EDG. With both trains blocked, the EDS is in compliance with GDC 17 and is analyzed.

It is understood that an accident of a different kind will exist if a degraded voltage condition occurs coincident with an accident (e.g., LOCA [versus the analyzed LOP + LOCA]). Should such an accident occur, the manual action described in the proposed ACTION statements could not be credited to protect the plant. However, the purpose of proposed ACTIONs 3.8.1.1.f and 3.8.1.1.g is to provide an appropriate response to degraded voltage prior to an accident by eliminating the malfunction of a different type (double sequencing) and an accident of a different type (e.g., degraded voltage + LOCA) for one train within one hour and for the second train within two hours. This duration of response is consistent with the required responses currently in the TSs 3.0.3, 3.8.2.1.a, and 3.8.1.1.a.

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

The margin of safety has not been reduced in that the train which has FBT blocked prior to the onset of an accident will be in conformance with GDC 17 (which is the basis to TS 3/4.8.1). Since the blocked train is in conformance with GDC 17 prior to the onset of an accident, it will support the single failure analyses and the safety analyses to the extent previously analyzed and licensed. The train not blocked will have its EDG started, loaded, and separated from offsite power prior to the end of the second hour. Action within two hours is consistent with TS 3.8.2.1.a. The proposed action recovers one train of A.C. sources in one hour and places the plant in a configuration of one less power source than is required by LCO 3.8.1.1 within two hours. Currently, TS ACTION 3.8.1.1.a (one power source inoperable) has a duration of seventy-two hours. The proposed ACTION requires responses within time frames consistent with TSs 3.0.3, 3.8.2.1.a, and 3.8.1.1.a, and therefore, does not reduce the margin of safety. Optionally, restoration of the second train by blocking FBT within two hours is also consistent with response times required by TS 3.0.3 and 3.8.2.1.a and therefore, also does not reduce the margin of safety. TS 3.8.1.1.a would not be required with both trains of FBT blocked as all four AC power sources would then be OPERABLE.

Regulatory Guide 1.93, ''Availability of Electric Power Sources,'' Revision 0, December 1974 recognizes that under certain conditions it may be safer to continue operation at full or reduced power for a limited time than to effect an immediate shutdown based on the loss of some of the required electric power sources. In an effort to minimize the risk to the health and safety of the public, the proposed ACTIONs 3.8.1.1.f and 3.8.1.1.g balance the risk of a forced shutdown against the risk of remaining at power with a degraded switchyard voltage.

Probabilistic Risk Analysis (PRA) has compared the probability of a core melt event for 1) blocking fast bus transfer in one train after one hour for the next seventy-one hours, and in the second train after two hours for the next seventy hours; 2) blocking fast bus transfer in one train after the first hour for the next seventy-one hours, and supplying power to the other train from the EDG after the second hour for seventy hours; and 3) a normal shutdown assuming the plant is in a normal configuration and no other transients or accidents except an uncomplicated reactor trip occurs during the shutdown process. Seventy-two hours was chosen for comparison purposes as the proposed ACTIONs would allow operation for up to seventy-two hours with one offsite circuit INOPERABLE.

The PRA has shown that the probability of a core melt event during power operation with FBT blocked in one train after one hour for the next seventy-one hours, and in the second train after two hours for the next seventy hours is approximately 1.91E-6. The probability of a core melt event during power operation with FBT blocked in one train after one hour for the next seventy-one hours and the EDG powering the opposite train after the second hour for the next seventy hours (the proposed configuration) is between approximately 1.91E-6 and 1.93E-6. A range is provided because the current PRA model can only model blocking both trains or the EDGs supplying both trains. The risk lies somewhere between the two values. The probability of a core melt event due to a normal shutdown assuming the plant is in a normal configuration and no other transients or accidents except an uncomplicated reactor trip occurs during the shutdown process is 2.4E-6. The risk can not be calculated for a forced shutdown with degraded switchyard voltage present but it is expected to be higher. Therefore, the analysis provided is conservative.

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

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

Date of amendment request: July 14, 1995

Description of amendment request: The proposed amendment would change the scram insertion times, Section 3.3.C, Minimum Critical Power Ratio section, Section 4.11.C and the associated bases in Section 2.1.1 and 3/4.3.

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:

Section 2.1 Bases - Safety Limits

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because equivalent fuel cladding protection (99.9 percent of all fuel rods do not experience transition boiling following a design basis transient) is provided.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not affect the function of any structure, system or component.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety because the utilization of current General Electric fuel designs provides an equivalent margin of safety. As stated previously, equivalent fuel cladding protection is provided and ensures that 99.9 percent of all fuel rods will not experience transition boiling following a design basis transient.

Section 3.3.C - Scram Insertion Times

1. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant increase in the probability of consequences of an accident previously evaluated. The correlation of the scram insertion times with the actual notch position will simplify the surveillance procedure while maintaining the accuracy of the test.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because no physical modifications are associated with the proposed change and it does not affect the function of any structure, system or component.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. The notch positions were chosen to coincide with the relative insertion values specified in the Technical Specifications. Use of the proposed combination of notch positions and scram insertion times will maintain the existing margins of safety that 99.9 percent of all fuel rods will not experience transition boiling following a design basis transient.

Section 4.11.C - Minimum Critical Power Ratio (MCPR) Calculation Method

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the method used to calculate the measured scram speed distribution is consistent with the PNPS [Pilgrim Nuclear Power Station] licensing basis.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not affect the function of any structure, system or component.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in the margin of safety because the proposed changes provide equivalent fuel cladding protection which ensures that 99.9 percent of all fuel rods will not experience transition boiling following a design basis transient.

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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Ledyard B. Marsh

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: September 17, 1993, as supplemented July 20, 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 September 17, 1993, and July 20, 1995, applications proposed to upgrade only Section 3/4.7 (Containment Systems) of the Dresden and Quad Cities TS.

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

The proposed 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 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 3/4.7 is based on STS guidelines or later operating BWR plants' NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has 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 Containment Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or 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. The additional surveillance requirements improve the reliability and availability of all affected systems and, therefore, reduce the consequences of any accident previously evaluated, as the probability of the systems outlined within Section 3/4.7 of the proposed Technical Specifications performing their intended function is increased by the additional surveillances.

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. Other changes 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. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, 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.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.7 is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden or Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. 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. 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 Containment 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 Containment 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.

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. Other changes 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 with 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 3/4.7 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or 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 or 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 or Quad Cities and maintain necessary levels of system or component reliability, 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 Containment 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

Connecticut Yankee Atomic Power Company, and Northeast Nuclear Energy Company, et al., Docket Nos.50-213, 50-245, 50-336, and 50-423 Haddam Neck Plant, and Millstone Nuclear Power Station, Units 1,2, and 3, Middlesex County and New London County, Connecticut

Date of amendment request: June 6, 1995

Description of amendment request: The proposed amendment will modify the size of the Plant Operations Review Committee (PORC) which will collectively have the experience and expertise in various areas of plant operation, and will clarify the composition of the Site Operations Review Committee (SORC).

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

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.

The PORC is an oversight group and helps to ensure that the units are operated in a safe manner. To accomplish this the PORCs provide their recommendations on the safety related activities to the Vice President - Haddam Neck Plant for Haddam Neck and to the respective Nuclear Unit Directors for Millstone. Each Millstone

Unit has its own PORC. It is proposed that the members of the

PORC be selected by the respective Nuclear Unit Director based on their knowledge and expertise in specific key plant functions. The Millstone Station has one SORC. The SORC is also an oversight group whose charter is to advise the Senior Vice President - Millstone Station on all matters related to nuclear safety at the Millstone site. The Haddam Neck Plant, being a single unit site, has one PORC, which advises the Vice President - Haddam Neck Plant. The members of the Haddam Neck Plant PORC will be selected by the Vice President - Haddam Neck Plant based on their knowledge and expertise in specific key plant functions. The PORC and SORC add to the defense-in-depth concept provided by the design, operation, maintenance, and quality oversight by promoting excellence through the conduct of their affairs and by maintaining a diligent watch over their responsibilities.

These administrative changes will revise the composition section of the technical specifications for the PORC members. Millstone Unit individuals will be appointed by the Nuclear Unit Directors if the individual meets one or more of the following areas of expertise: Plant Operations, Engineering, Reactor Engineering, Maintenance, Instrumentation and Controls, Health Physics, Chemistry, Work Planning and Control, and Quality Services. The Haddam Neck Plant, due to its broader scope of review also include[s] an individual experienced in Security and specific experience in Electrical Maintenance and Mechanical Maintenance. The individuals who will serve on PORC shall continue to meet the criteria of ANSI N18.1-1971. This approach is consistent with the standard technical specifications and NUREG 0800, Section 13.4. For SORC at the Millstone Station, the method of identifying who shall serve as Vice Chairperson has been modified for clarity. The Site Services Director position is proposed to be eliminated since this position no longer exists. The functions previously performed by this individual have been assumed by those individuals who currently serve on SORC. Finally, [the TS relating to] the individual who shall represent Quality and Assessment Services shall be modified to allow a qualified member of Quality and Assessment Services to serve on SORC.

The remaining portions of the technical specifications related to PORC and SORC are not being revised.

These modifications broaden the unit committee participation and reflect current organizational positions and will not increase the probability of occurrence or the 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 administrative enhancements to the composition of the PORC and Millstone Station SORC will not affect the way in which the units are physically operated. These administrative changes to PORC and SORC continue to meet the guidelines of ANSI N18.7-1976. The modifications to PORC and SORC continue to allow these groups to provide a thorough review of activities at the units.

The proposed modification does not impact any initiating events, and, therefore, cannot create the possibility of any new or different kind of accident from any accident previously evaluated.

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

These proposed administrative changes will not impact the margin of safety provided by PORC and SORC. The PORC and SORC will continue to be staffed by qualified individuals experienced in the operation of the plants. These administrative changes will modify how the composition of the PORC and SORC members are presented in the technical specifications, but will not adversely impact their ability to review and comment on operations at the units.

These changes do not impact any protective boundaries nor do they impact the safety limits for the protective boundaries. These proposed changes are administrative in nature. Therefore, there is no 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: Russell Library, 123 Broad Street Middletown, Copnnecticut 06457, for the Haddam Neck Plant, and the Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, for Millstone 1, 2, and 3.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: July 5, 1995

Description of amendment request: The proposed amendment would change the Administrative Controls section of the Palisades Technical Specifications. The changes involve deleting training requirements in the Administrative Controls section, revising the Plant Review Committee composition, and revising the function and composition of the plant safety and licensing staff review requirements.

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

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

This change does not affect the probability or consequences of an accident. The changes are administrative, deleting an unnecessary specification on staff training requirements, eliminating the specific references to the Nuclear Engineering and Construction Organization (NECO) staff, and requiring that the Plant Review Committee (PRC) chairman, alternate chairman, and members be designated in administrative procedures by the Plant General Manager. Further administrative changes clarify the function of the Plant Safety and Licensing organization and eliminate the numerical requirement for five staff members to fulfill the organization function.

The removal of an obsolete staff training requirement does not diminish the regulatory requirement to have an adequately trained staff. The accredited training programs for the plant staff ensure an appropriate level of training is conducted to maintain an appropriate skill and knowledge base for the staff. The requirements of 10CFR55 provide the necessary rules for operator licenses. Since a trained staff will be maintained, there will [be] no increase in the probability or consequences of an accident as a result of this change.

The composition of the PRC will not be affected by this change as it will, at a minimum, be comprised of personnel from the operations, engineering, radiological services and maintenance departments as required by the Technical Specifications. The composition of the Plant Safety and Licensing organization as a whole may change. The function of the organization as it relates to these Technical Specifications, however, will not be affected. These changes have no affect on the plant accident analyses. Qualified personnel will continue to conduct the PRC and Plant Safety and Licensing reviews. Therefore, the changes do not increase the probability or consequences of an accident.

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

The proposed changes are administrative and do not create the possibility of a new or different kind of accident. Staff training will continue to meet the accreditation requirements of the National Academy for Nuclear Training Accreditation Board and the requirements for the Systematic Approach to Training. Operators' license training will continue to meet the regulatory requirements of 10CFR55. Activities conducted by the Plant Review Committee and the Plant Safety and Licensing staff will continue to be accomplished by a staff which meets the qualification requirements of the Technical Specifications. These administrative changes will not affect the operation of the plant or the safety function of plant equipment nor will it affect the quality of the review activities. Therefore, there will be no possibility that a new or different kind of accident will be created.

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

The changes do not affect installed plant equipment nor do they affect plant operations. These administrative changes have not affected the probability or consequences of a previously analyzed accident or created the possibility of a new or different kind [of] accident from any previously evaluated. Therefore, they do 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 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: Van Wylen Library, Hope College, Holland, Michigan 49423.

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

NRC Project Director: John N. Hannon

Duquesne Light Company, et al., Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: October 11, 1994, as supplemented June 23, 1995.

Description of amendment request: The proposed amendments would revise Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2) Technical Specifications (TSs) 1.18, ''Quadrant Power Tilt Ratio,'' 3/4.2.4, ''Quadrant Power Tilt Ratio,'' the Table Notation of TS Table 3.3.-1, ''Reactor Trip System Instrumentation,'' and associated Bases to incorporate the guidance provided in the NRC's Improved Standard Technical Specifications (NUREG-1431) applicable to these TSs. The proposed amendments would clarify the requirements of the subject TSs with regard to the use of excore power range neutron flux detectors to monitor quadrant power tilt ratio when an excore power range neutron flux instrument is inoperable. The proposed change would also make several minor editorial changes in the subject TSs.

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

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

The existing quadrant power tilt ratio (QPTR) definition and Surveillance Requirement (SR) 4.2.4.c are inconsistent concerning reactor power limitations when performing QPTR surveillance requirements. The proposed change modifies these and related requirements to improve the understanding and consistency by generally incorporating the Improved Standard Technical Specification (ISTS) requirements of NUREG-1431.

Editorial changes have been incorporated throughout the proposed specifications to address ISTS or plant specific convention and do not affect the accident analyses. The QPTR definition has been modified to reflect the ISTS wording and eliminate the inconsistency with SR 4.2.4.c. This change does not reduce the QPTR testing requirements or affect the accident analyses assumptions. The current action statements require power reduction along with a reduction in power range high neutron flux trip setpoints when the QPTR exceeds the limit. This ensures the core conditions are consistent with the accident analyses assumptions. With the modified action statements and the QPTR exceeding the limit, power reduction is also required along with performing a flux map to verify the peaking factors are within the accident analyses assumptions. In addition, the safety analyses must be re-evaluated to confirm the results remain valid prior to increasing power with an indicated tilt condition. The new action statements provide methods different from the current requirements. However, they satisfy the same objective, to ensure the conditions assumed in the accident analyses are maintained. Therefore, these changes will not involve significant increase in the probability or consequences of an accident previously evaluated.

The current surveillance requirements define the methods and frequencies for verifying the QPTR is within the limit specified in the limiting condition for operation. The proposed SRs include associated notes that allow separation of a power range channel into two portions made-up of the Nuclear Instrumentation System (NIS) and the excore detector portion. If an excore detector portion of a power range channel is inoperable, then the power range channel is inoperable since the detector provides input to the NIS which inputs to the solid state protection system. However, if the excore detector is operable and the NIS is inoperable, then the power range channel is inoperable but the ability to monitor the QPTR is unaffected. When the NIS portion of a channel is inoperable, appropriate actions are applied in accordance with Specification 3.3.1. The new SRs continue to require the same testing and frequencies as the current SRs along with reducing the need to interpret the requirements when special conditions exist. Therefore, the proposed SRs will not affect the accident analyses or significantly increase the probability or consequences of an accident previously evaluated.

Table 3.3-1 Action 2 applies when a power range channel is inoperable. This action has been reformatted to incorporate changes similar to those adopted in the QTPR SR which allow separation of a power range channel into the NIS portion and the excore detector portion. Proposed Action 2.a applies to an inoperable power range high neutron flux channel and Action 2.b applies to ''all other channels'' which includes the Low Setpoint function along with the High Positive and High Negative Rate functions. The new action is modified by Note (3) to allow bypassing the inoperable channel for surveillance testing and setpoint adjustment and by Note (4) that only requires performing SR 4.2.4 when the power range high neutron flux channel input to QPTR is inoperable. The new action does not require reducing the power range neutron flux setpoint like the current action since the proposed action is to perform the QPTR surveillance or shutdown which is more conservative than the current action requirement, otherwise, the new action requires essentially the same steps to be performed as the current action. Therefore, the proposed action will not affect the accident analyses or involve a significant increase in the probability or consequences of an accident previously evaluated.

These changes are proposed to allow flexibility in plant operations by modifying the QPTR action and surveillance requirements to allow separation of a power range channel into the NIS portion and the excore detector portion. The modified action and surveillance requirements continue to provide monitoring of those parameters required to ensure the core is operating safely. Since these changes are not significantly different from the current requirements and no change is being introduced that would affect the accident analyses assumptions, we have concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes incorporate modifications generally consistent with the ISTS QPTR requirements to ensure the core power distribution is adequately monitored. The revised action statements provide for peaking factor verification as a logical compensatory measure to ensure the core is operating within required limits. This is more conservative than the current requirements and provides additional assurance that Specification 3.2.4 will continue to govern the QPTR limitations in a manner consistent with the accident analyses assumptions. The revised SR provides clear and understandable testing requirements to reduce confusion concerning how the QPTR is to be monitored based on plant conditions. The proposed change does not introduce any new mode of plant operation or require any physical modification to the plant, therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The QPTR limit ensures that the gross radial power distribution is maintained within the assumptions used in the safety analyses. The QPTR is one of the variables that is monitored to ensure the core operates within the bounds used in the safety analyses. When the QPTR is maintained below 1.02 it provides an indication that the peaking factors are within the limiting values by preventing and undetected change in the gross radial power distribution. The proposed changes ensure the required parameters are verified during the applicable conditions and on a consistent basis, therefore, these changes will not reduce the margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: May 19, 1995

Description of amendment request: The proposed amendments revise the specifications to permit the reactor building personnel airlock doors to remain open during fuel handling.

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

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

The proposed change would allow the containment personnel airlock doors to remain open during fuel movement and core alterations. These doors are normally closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident. These doors are not initiators of any accident. The probability of a fuel handling accident is unaffected by the position of the containment personnel airlock doors.

The proposed change alters assumptions made in evaluating the radiological consequences of a fuel handling accident inside the reactor containment building. Allowing the containment personnel airlock doors to remain open during fuel movement and core alterations does increase, however not significantly, the consequences of a fuel handling accident inside containment. Previously, the fuel handling accident inside containment was bounded by the fuel handling accident analysis in the spent fuel pool area of the auxiliary building. Part of the dose increase has been offset by the increase in the minimum decay time before irradiated fuel may be moved inside the reactor containment building. Extending the minimum decay time actually decreases the consequences of a fuel handling accident by reducing the radioactive inventory of the irradiated fuel which could possibly be released during a fuel handling accident. The revised fuel handling accident analysis results in maximum offsite doses of 43.4 Rem and 41.8 Rem to the thyroid and 0.616 Rem and 0.598 Rem to the whole body for ANO-1 and ANO-2, respectively. The calculated offsite doses are well within the limits of 10CFR Part 100. Also, the calculated doses are larger than the actual doses which would be expected during a fuel handling accident because the calculation does not incorporate the closing of at least one of the personnel airlock doors following evacuation of containment. The proposed change would significantly reduce the dose to workers in the containment in the event of a fuel handling accident by expediting the containment evacuation process.

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

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

The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design, configuration, or method of operation of the plant.

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

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

This proposed change has the potential for an increased dose at the site boundary due to a fuel handling accident; however, the dose remains within acceptable limits. The margin of safety as defined by 10CFR Part 100 has not been significantly reduced. There is an increase in the calculated offsite dose resulting from a fuel handling accident; however, the increase is not significant and is well within the limits specified in 10 CFR Part 100. The overall significance will be offset by the increased minimum decay time, the decreased potential radiation dose to workers, and the increased availability of the personnel airlock door in the event of a fuel handling accident. Closing at least one of the personnel airlock doors following an evacuation of containment, further reduces the offsite doses in the event of a fuel handling accident which partially compensates for the higher offsite doses calculated as a result of this proposed change.

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

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

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

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

NRC Project Director: William D. BecknerEntergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: March 17, 1995

Description of amendment request: The proposed amendment revises requirements associated with channel functional tests of the core protection calculator following a high temperature alarm.

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

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

The core protection calculators (CPCs) are not accident initiators, therefore this change does not increase the probability of an accident previously evaluated.

The core protection calculators (CPCs) are dedicated minicomputers that receive key parameters necessary to calculate the departure from nucleate boiling ratio (DNBR) and local power density (LPD) and issue a reactor trip command prior to reaching plant conditions that may damage the fuel in the reactor. Subjecting a computer to elevated temperatures may affect the reliability of the computer calculations. This change in the Arkansas Nuclear One-Unit 2 (ANO-2) Technical Specifications (TS) will require a verification of the CPC operability, by the performance of a channel functional test, in the event a cabinet high temperature switch is actuated. This is a more accurate indication of the operating environment of the CPCs than the current requirement to perform the test based upon room temperature. The ability of the CPCs to monitor DNBR and LPD and issue a trip command when appropriate will not be affected in any way by this change, therefore the consequences of an accident previously evaluated are not increased.

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

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

Because the proposed changes do not alter the design, configuration, or method of operation of the plant, they do not create the possibility of a new or different kind of accident from any previously evaluated.

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

These proposed changes do not alter the acceptance criteria of any surveillance requirements. The changes do not alter any assumptions used in accident analysis, change any actuation setpoints, nor allow operations in any configuration not previously analyzed. This change will trigger a verification of affected CPC operability based on cabinet temperature instead of room temperature, which is a more accurate indication of the operating environment of the CPC computer. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: April 4, 1995

Description of amendment request: The proposed amendment revises operating criteria and requirements associated with containment personnel air locks.

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

Criterion 1 - Does Not Involve a Significant Increase in the

Probability or Consequences of an Accident Previously Evaluated.

The containment air locks are passive components integral to the containment structure and are not evaluated to be accident initiators, therefore, the proposed amendment does not involve an increase in the probability of an accident previously evaluated.

Each air lock door is rated for and tested to full design pressure of the containment building. If one door were inoperable in each air lock, the remaining door, since required to remain closed and locked, would provide the necessary fission product barrier to prevent an uncontrolled release, therefore the amendment allowance for an inoperable air lock door in each air lock does not increase the consequences of any previously evaluated accident.

During a situation where one containment air lock door is inoperable and the operable door is opened, a breech in containment integrity would essentially exist while the operable door remains open. The time required for a containment air lock door to be open for ingress or egress does not exceed two to three minutes. The amendment provision to allow unlocking and opening an operable air lock door for ingress and egress to facilitate air lock maintenance necessary to restore operability does not increase the consequences of any previously evaluated accident since the time necessary for the door to be open is bounded by the existing one hour time allowance for an actual breech of containment integrity (TS 3.6.1.1.)

The containment air lock interlock functions to prevent simultaneous opening of both air lock doors thereby creating a breech in containment integrity. A dedicated individual stationed at the air lock to administratively control door operations, or locking closed an operable door will adequately assure containment integrity. The addition of this technical specification action statement, therefore, does not increase the consequences of any previously evaluated accident.

Performance of the overall air lock leakage test requires opening the outer air lock door for installation of the mechanical dogging devices on the inner door. The current technical specifications make no provisions for this entry and thus would require a plant shutdown if the inner door was inoperable in an air lock. The proposed amendment removes the requirement to shut down when the barrel leak rate is due. The time required for the containment air lock doors to be opened for dog installation would be the same as for ingress and egress as discussed above, therefore this change does not increase the consequences of any previously evaluated accident.

10 CFR 50, Appendix J contains containment leakage testing requirements, including specific requirements for containment building air locks. Changing the TS surveillance requirements to refer to 10 CFR 50, Appendix J for these test requirements will not degrade these tests, therefore this change does not increase the consequences of any previously evaluated accident.

The air lock door seal pressure test is performed any time the air lock is used for containment access during modes of operation when containment integrity is required. The door seal test is intended to be a gross test to verify that the door seals were not damaged during the opening and closing cycle(s). This test does not replace the required overall barrel leakage test. Based on information provided by the air lock vendor, a test pressure of 10 psig is sufficient to perform this gross seal verification. A change in the allowable leakage rate is requested to remove a specific numerical value from the TS surveillance requirements section and replace it with a fraction of L_{aG}. This new acceptable leakage rate remains relatively insignificant and is bounded by the overall air lock leakage rate. Based on these facts this change in test pressure and associated acceptance criteria does not increase the consequences of any previously evaluated accident.

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

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

Because the proposed changes do not change the design, configuration, or method of operation of the plant, they do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes to ANO-2 TS involve allowing brief breaches in containment integrity for the purpose of repairing inoperable air lock components or performing surveillances required by 10 CFR 50, Appendix J. These cases are adequately bounded by the one hour allowable outage time afforded by TS 3.6.1.1.

The addition of a specific action statement addressing an inoperable air lock interlock provides those actions necessary to assure the maintenance of containment integrity. This is achieved by locking an operable door in the affected air lock when not in use and stationing a dedicated individual at the air lock, during periods of ingress and egress, whose sole responsibility is to insure only one air lock door is opened at a time thereby duplicating the function of the mechanical interlock.

The proposed changes also consist of administrative changes removing an outdated exemption to 10 CFR 50, Appendix J and removing specific surveillance requirements from the specifications, instead referring to the controlling requirements of 10 CFR 50, Appendix J. This is consistent with the provisions of NUREG 1432 ''Revised Standard Technical Specifications for Combustion Engineering Plants,'' Rev. 0.

None of the proposed changes increase the allowable overall air lock leakage rate, nor affect the acceptance criteria of the overall integrated containment leakage rate. All of the changes are bounded by existing analyses for all evaluated accidents and do not create any situations that alter the assumptions used in these analyses. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: May 19, 1995

Description of amendment request: The proposed amendment adds criteria to address optional inspections of steam generator tubes.

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

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

Steam generator tubes are inspected on a periodic basis to reduce the probability of a steam generator tube rupture or tube leakage. Five special interest groups are being added for optional inspections in addition to the general tube inspections currently required by the technical specifications. These special interest groups define areas of tubes where known or potential degradation mechanisms may exist for which additional inspection, above that currently required in the technical specifications, may be beneficial. Inspection of these special interest groups may utilize probes which more readily detect indications which may be found in the special interest areas. The increased detection capability will reduce the probability that a structurally significant flaw will go undetected during an inspection. The minimum sample size and expansion criteria (should a flaw be found) for inspections of special interest groups are based on percentages of tubes potentially affected by the specific degradation mechanisms for which the special inspection is being performed. The percentages used are the same as used for the current general tube inspections. The expansion criteria allow expansion within the area of interest without affecting the expansions of any general tube inspection. By expanding within the area of interest, a more complete inspection for the defects caused by a specific degradation mechanism can be performed than if the expansion were conducted in tubes not necessarily affected by the degradation mechanism, which is possible with the current technical specifications. Therefore, this change does not involve a significant increase in the probability of an accident previously considered.

The proposed change does not increase the amount of radioactive material available for release or modify any systems used for mitigation of such releases during accident conditions. The steam generator tubing will continue to be examined on the frequency currently specified in the technical specifications. This change will allow steam generator examinations to focus on known areas of interest without requiring unnecessary expansion. The integrity of the steam generators will continue to be assured at an equivalent level. Therefore, the change does not involve a significant increase in the consequences of any accident previously evaluated.

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

Special inspections such as the ones being added to the technical specifications have been conducted in the past at ANO-2. The method of inspection, pushing or pulling a probe through the steam generator tubes from the primary side, is the same method employed for the current technical specification required inspections. Inspection methodology is not being changed by incorporation of these special interest groups into the technical specifications. No design or operational characteristics of the plant are changed by the proposed amendment.

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

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

The proposed amendment adds special interest groups for optional inspection into the technical specifications. These inspections concentrate on areas of interest using inspection methodology that is equivalent or better at finding specific types of flaws than the methodology used for the currently required general tube inspections. If the special interest groups are not inspected, the existing technical specification requirements for inspection still apply.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: May 19, 1995

Description of amendment request: The proposed amendment increases the allowed outage time for an emergency diesel generator from 72 hours to seven days. Additionally, the amendment authorizes one, ten-day diesel generator maintenance outage every fuel cycle.

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

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

The emergency diesel generators (EDGs) are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. EDGs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The EDGs provide backup power to components that mitigate the consequences of accidents. The proposed changes to allowed outage times (AOTs) do not affect any of the assumptions used in deterministic safety analysis.

In order to fully evaluate the EDG AOT extension, probabilistic safety analysis methods were utilized. The results of these analyses indicate no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD-996, Combustion Engineering Owners Group ''Joint Applications Report for Emergency Diesel Generators AOT Extension.''

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

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

This proposed change does not alter the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the technical specification limiting conditions for operation or their bases which support the deterministic analyses used to establish the margin of safety. Evaluations used to support the requested technical specification changes have been demonstrated to be either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD-996.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: May 19, 1995

Description of amendment request: The proposed amendment increases the allowed outage time for an inoperable

Safety Injection Tank (SIT) from one hour to 24 hours. Additionally, the amendment limits power operation to 72 hours when certain SIT related instrument functions are inoperable.

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

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

The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System. The SITs are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

SITs were designed to mitigate the consequences of Loss of Coolant Accidents (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Therefore, the consequences of accidents previously evaluated do not change.

In order to fully evaluate the effect of the SIT Allowable Outage Time (AOT) extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD-994, Combustion Engineering Owners Group ''Joint Applications Report for Safety Injection Tank AOT/STI Extension.''

The change pertaining to SIT inoperability based solely on instrumentation malfunction does not involve a significant increase in the consequences of an accident as evaluated and endorsed by the NRC in NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements.''

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

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

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrated that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD-994.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: May 19, 1995

Description of amendment request: The proposed amendment increases the allowed outage time for one train of low pressure safety injection from 72 hours to seven 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:

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

The low pressure safety injection system (LPSI) is part of the Emergency Core Cooling System subsystem. Inoperable LPSI components are not considered to be accident initiators. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The LPSI system was designed to mitigate the consequences of a large loss of coolant accident (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis.

In order to fully evaluate the LPSI AOT extension, probabilistic safety analysis methods were utilized. The results of these analyses indicate no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD-995, Combustion Engineering Owners Group ''Joint Applications Report for Low Pressure Safety Injection System AOT Extension.''

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

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

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the technical specification limiting conditions for operation or their bases which support the deterministic analyses used to establish the margin of safety. Probabilistic evaluations used to support the requested technical specification changes have been demonstrated to be either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD-995.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: June 26, 1995

Description of amendment request: The amendment revises the snubber visual inspection intervals to match the schedule developed by the NRC staff for use with a 24 month refueling interval. This schedule was documented in Generic Letter 90-09. The licensee has made wording changes not contained in Generic Letter 90-09. These changes are as follows:

a) Section 4.5.Q.1 - GL 90-09 wording ''...performance of the following augmented inservice inspection program in addition to the requirements of Section 4.0.5.''

Proposed Technical Specification wording ''...performance of the following inspection program.''

b) Section 4.5.Q.1.a - GL 90-09 wording ''...based on the criteria of Table 4.7.2 and the first inspection interval determined using the criteria shall be based upon the previous inspection interval established by the requirements in effect before Amendment (*). ''Proposed Technical Specification wording ''...based on the criteria provided in Table 4.5.1.''

c) Section 4.5.Q.1.b - GL 90-09 wording ''...All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval.''

Proposed Technical Specification deletes this sentence.

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

The proposed amendment would revise the basis for the snubber visual inspection to be consistent with the bases described in Generic Letter 90-09.

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

The proposed change does not affect the probability of occurrence nor does it affect the consequences of an accident previously evaluated as the requested visual inspection interval has been determined generically to be a safe and acceptable alternative to the existing visual inspection requirements as documented by the NRC in Generic Letter 90-09. With the completion of over 25 years of operating experience and only detecting one visual inspection failure, GPU Nuclear agrees that the existing intervals are overly conservative and can be extended to those described in the generic letter.

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

As the requested change deals only with the frequency of visual inspection and not with the content, scope, or acceptance criteria of the inspection, no new or different type of accident has been created.

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

The margin of safety as defined in the bases of the Technical Specifications is not reduced as the requested requirements provide the same degree of confidence in snubber operability at the existing requirements.

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

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

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

NRC Project Director: Phillip F. McKee

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 31, 1995

Description of amendment request: The proposed amendment would modify (by relocation to the Technical Requirements Manual) Technical Specification (TS) 3/4.1.2.1, Boration Systems/Flow Paths - Shutdown, TS 3/4.1.2.2, Boration Systems/Flow Paths - Operating, TS 3/4.1.2.3, Charging Pumps - Shutdown, TS 3/4.1.2.4, Charging Pumps - Operating, TS 3/4.1.2.5, Borated Water Sources - Shutdown, TS 3/4.1.2.6, Borated Water Sources - Operating, TS 3/4.4.2.1, Safety Valves - Shutdown, and the associated Bases.

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

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

The proposed change to the subject Technical Specifications is of an administrative nature in that the subject Technical Specifications and Bases will be relocated in their entirety to the Technical Requirements Manual. Future changes to the relocated requirements will be in accordance with 10CFR50.59 and approved station procedures.

Whether the listed Technical Specifications and Bases are located in Technical Specifications or the Technical Requirements Manual has no effect on the probability or consequences of an accident previously evaluated.

The proposed change does not alter the assumptions previously made in the listed Technical Specifications. The proposed change allows the Commission and the South Texas Project more effective use of personnel resources to control requirements that meet the four Criteria in the Final Policy Statement. The proposed change will not change the dose to workers.

Since the probability of an accident is unaffected by administratively relocating the subject Technical Specification, and the doses are not affected and do not exceed acceptance limits, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the subject Technical Specifications is of an administrative nature in that the subject Technical Specifications and Bases will be relocated in their entirety to the Technical Requirements Manual. Future changes to the relocated requirements will be in accordance with 10CFR 50.59 and approved station procedures. Whether the listed Technical Specifications and Bases are located in Technical Specifications or the Technical Requirements Manual has no effect on any previously evaluated accident. It does not represent a change in the configuration or operation of the plant and, therefore, does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

The proposed change to the subject Technical Specifications is of an administrative nature in that the subject Technical Specifications and Bases will be relocated in their entirety to the Technical Requirements Manual. Future changes to the relocated requirements will be in accordance with 10CFR50.59 and approved station procedures. The margin of safety is not reduced when the requirements are relocated to a Licensee-controlled document because the requirements to change a License Basis Document via the 10CFR50.59 process ensure the same questions concerning the margin of safety required for license amendments are asked. Therefore, this proposed change does not significantly reduce the margin of safety.

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

Local Public Document Room location: Wharton County Junior College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

NRC Project Director: William D. Beckner

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: June 28, 1995

Description of amendment request: The proposed amendment would revise technical specifications related to the standby liquid control (SLC) system. The proposed changes include increasing the required reactor pressure vessel boron concentration and modifying the SLC pump operability testing surveillance frequency from monthly to quarterly.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The current analysis requires the SLC system to be capable of bringing the reactor 3% delta - k subcritical assuming a cold xenon free condition. The increase in SLC storage tank boron concentration limits will ensure this capability is maintained for future reload cores using the same 3% delta - k shutdown reactivity margin without imposing restrictions in cycle exposure for current and future anticipated core configurations. The change in the surveillance frequency for SLC pump operability testing to once each three months is in agreement with the ASME Code. The relaxation of the testing interval for the SLC pumps decreases pump degradation, and eliminates an unnecessary burden on personnel resources without compromising plant safety. In addition, the administrative changes only correct typographical and editorial errors.

Since these proposed changes do not affect precursors for any accident or transient analyzed in Chapter 14 of the USAR, there is no increase in the probability of any accident previously evaluated. Furthermore, since these changes will ensure the ability of the SLC system to mitigate the consequences of an accident for future anticipated core designs, they do not involve a significant increase in the consequences of any accident previously evaluated.

2. The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. The change in the SLC storage tank boron concentration limits will ensure that a cold xenon-free reload core can be brought to a subcritical condition as previously analyzed. The change in the frequency of the SLC pump operability testing to once each three months is in agreement with the ASME Code. The relaxation in the testing interval for the SLC pumps decreases pump degradation, and eliminates an unnecessary burden on personnel resources without compromising plant safety. In addition, the administrative changes only correct typographical and editorial errors.

These proposed changes do not affect the design, function, or operation of the SLC or any other system. Also, these changes do not introduce any new modes of operation or modify existing equipment design. Therefore, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes will not create a significant reduction in the margin of safety. The proposed increase in the required boron concentration in the reactor pressure vessel will ensure the SLC system will be capable of bringing a cold xenon-free reload core subcritical while maintaining the 3% delta - k shutdown reactivity margin as specified in the previous operating cycle. The change in the frequency of SLC pump operability testing to once each three months is in agreement with the ASME Code. The relaxation in the testing interval for the SLC pumps decreases pump degradation, and eliminates an unnecessary burden on personnel resources without compromising plant safety. In fact, it increases SLC system availability. In addition, the administrative changes only correct typographical and editorial errors. Therefore, it is concluded that the requested changes do not create a significant reduction in the existing margin of safety as defined in the Technical Specifications.

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: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68602-0499

NRC Project Director: William D. Beckner North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: June 16, 1995

Description of amendment request: The proposed amendment would change the minimum boron concentration specified for the refueling water storage tank (RWST) in Limiting Condition for Operation (LCO) in Technical Specification (TS) 3.1.2.5 and would replace the minimum specified concentration for boron with an acceptable range of boron concentration for the RWST and the accumulators in the LCOs for TS 3.1.2.6, 3.5.1.1, and 3.5.4.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the changes are proposed to assure that the post-event shutdown margin required by the Technical Specifications will continue to be met and the consequences of a boron dilution event will remain as previously evaluated. The changes do not affect the design or manner of operation of any structure, system, or component important to safety.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the manner by which the facility is operated and do not involve a change to any structure, system, or component important to safety. The proposed changes merely assure that station will be operated within original design limits.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes merely assure that the station will continue to be operated within the original design limits. Therefore, the acceptance criteria for previously evaluated accidents will continue to be met.

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: Thomas Dignan, Esquire, Ropes & Gray, One International Place, Boston MA 02110-2624.

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 11, 1995

Description of amendment request: The proposed amendment modifies Technical Specification 3.5.F.7 to also allow the use of pull-to-lock switches to defeat the automatic initiation of the emergency core cooling system (ECCS) while in the refuel condition. The proposed amendment also makes administrative changes and makes changes to the associated Bases section.

*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:

NNECO has reviewed the proposed change in accordance with 10 CFR 50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed change does not involve an SHC because the changes would not:

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

This change to LCO [Limiting Condition for Operation] 3.5.F.7(e) will allow an alternative means of de-energizing power to the selected ECCS pump motors during refueling. The current technical specification already allows these motors to be de-energized. Use of the pull-to-lock switches provides a safer method of achieving this condition. The pull-to-lock condition of the switches is annunciated in the control room. Therefore, the switches will not be inadvertently left in the pull-to-lock position.

Deletion of the statement that the 4160 volt supply breakers are racked in does not affect the requirement of LCO 3.5.F.7 to ensure the specified ECCS subsystems are OPERABLE.

Therefore, there is no change in the probability or consequences of an accident previously analyzed due to this change.

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

The use of an alternative means of de-energizing power from the selected ECCS pump motors does not create a possibility of a new or different kind of accident. Using the control room pull-to-lock switch to disable the pump motor circuit breaker has the same effect on the ECCS pump as the removal of the circuit breaker from the switchgear.

Deletion of the statement that the 4160 volt supply breakers are racked in does not affect the requirement of LCO 3.5.F.7 to ensure the specified ECCS subsystems are OPERABLE.

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

The proposed change to the Millstone Unit No. 1 Technical Specifications does not reduce the margin of safety. By using the control room pull-to-lock switches to disable the ECCS pump motors, instead of racking out the pump motor circuit breakers, it is possible to reenergize the ECCS pumps more quickly in an emergency, should one occur. The time savings can be translated into added safety margin from a shutdown risk perspective. The ability to disable and enable the pumps from the control room, instead of the switchgear area, also contributes to this added safety margin.

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: Ms. L. M. Cuoco, Senior Nuclear Counsel, 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 18, 1995

Description of amendment request: The proposed amendment request will add operability and surveillance requirements for reactor pressure vessel (RPV) overfill protection instrumentation. The proposed amendment will also add the associated Bases.

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

NNECO has reviewed the proposed change in accordance with 10 CFR 50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed change does not involve an SHC because the change would not:

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

The new LCO [Limiting Condition for Operation] and surveillance requirements ensure that the reactor high water level feedwater pump trip instrumentation is available. This technical specification change does not involve the addition of new equipment or logic. This change does not add new surveillance requirements for the instrumentation. This change simply establishes requirements for the operation and surveillance of reactor high water level feedwater pump trip instrumentation in the technical specifications. The implementation of this technical specification change will decrease the likelihood of an RPV overfill. No other postulated event is affected by the addition of this instrumentation to the technical specifications.

Thus, adding the proposed requirements to the technical specifications will not increase the probability or consequences of any previously evaluated transients or accidents.

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

No new failure modes are introduced by the addition of the reactor high water level feedwater pump trip instrumentation LCO and surveillance requirements. Modifying the technical specifications to formally add surveillance requirements already being performed in accordance with plant procedures will not modify plant response to any operational or transient event. Increasing the surveillance interval of the LITS [level indicating transmitter switches] from annual to once per operating cycle will not significantly affect reliability. Ensuring the operability of installed instrumentation does not add new or different kinds of accidents.

Therefore, the new LCO and surveillance requirements do not create the possibility of a new or different kind of accident.

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

The surveillance requirements being added in this change are consistent with current surveillances being performed for this instrumentation, with the exception that the LITS are currently calibrated on an annual rather than operating cycle basis. These surveillance and shutdown requirements ensure that protection from RPV overfill is maintained as assumed in the safety analyses.

Therefore, there is no impact on 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: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: July 7, 1995

Description of amendment request: The proposed change to technical specification 3/4.7.6 is being made to: 1) increase the allowable control room air conditioning (CRAC) system in-leakage from 100 cubic feet per minute (cfm) to 130 cfm; 2) provide a more conservative value for the maximum differential pressure across the high efficiency particulate air (HEPA) filters and charcoal adsorbers; 3) clarify that when the CRAC system is shifted to ''recirculation,'' this will be performed from the normal mode; and 4) modify the corresponding basis to reflect the above changes and to note that there are certain infrequent situations during which the control room emergency ventilation system (CREVS) will not automatically operate.

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

...The proposed changes do not involve an SHC because the changes will not:

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

The CRAC system in the recirculation mode is used to mitigate the effects of an accident. Surveillance Requirement 4.7.6.1.e.2 has been modified to clarify that the system will automatically switch from the normal mode into a recirculation mode. This change and the proposed modifications to the acceptance criterion for the differential pressure across the HEPA filters and charcoal adsorbers and the increase in the control room in-leakage have no [e]ffect on the probability of an accident previously evaluated. The consequences of the accidents that have been previously evaluated have been reviewed to determine the impact of these proposed modifications. The increase in the in-leakage will affect the results of previously generated accident analysis. The accidents evaluated, namely the Millstone Unit No. 1 MSLB [main steam line break] and LOCA [loss-of-coolant accident], Millstone Unit No. 2 LOCA, both high and low wind speed case, and Millstone Unit No. 3 LOCA have been reviewed. The Millstone Unit No. 1 LOCA doses to the Millstone Unit No. 2 control room were qualitatively determined to be bounded by the Millstone Unit No. 2 LOCA cases. Therefore the Millstone Unit No. 1 LOCA was not performed. The remaining accidents were performed. The resultant doses are nearly identical to the existing doses found in the Millstone Unit No. 2 Final Safety Analysis Report and are all within the regulatory limits. To perform these revised control room dose calculations, NNECO used certain new assumptions which NNECO believes better model the control room and the effects the accident will have on the control room. The most significant change with the assumptions is the use of ICRP 30 in lieu of Regulatory Guide 1.109, Revision 1 for iodine dose conversion factors. The NRC has used ICRP 30 over the past 5 years for other applications and its use in this instance is appropriate.

The change in the acceptance criterion for the differential pressure across the HEPA filter and charcoal adsorbers is a conservative modification in that the value given is a plant specific value and will be more indicative of blocked or clogged filters in actual plant conditions. These proposed changes do not have any negative impact on 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 modifications to Surveillance Requirement 4.7.6.1 will clarify a portion of a surveillance requirement and will modify the differential pressure across the HEPA filters and the charcoal adsorbers. These changes will not create the possibility of a new or different kind of accident from any previously evaluated. The increase in the allowable control room in-leakage value from it[s] current level of 100 cfm to its new value of 130 cfm also does not create the possibility of a new or different kind of accident. The CRAC system is used to mitigate the consequences of an accident.

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

The proposed modifications do not decrease the margin of safety provided. Using the new accident assumptions, the limiting accidents were re-calculated to determine the impact on the Millstone Unit No. 2 control room. These values are similar to the values found in the Millstone Unit No. 2 Final Safety Analysis Report and the Millstone Unit No. 2 Safety Evaluation Report and are within the regulatory limits established for the control room operators. Since the re-calculated doses have been shown to be within limits, it has been concluded that there is no 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 Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 8, 1995

Description of amendment request: The Millstone Unit No. 3 Technical Specification Section 3/4.8.4.3 requires removal of electrical power to the safety injection accumulator isolation valves in Modes 1, 2, 3, and 4 in order to protect the containment electrical penetrations and penetration conductors. Bases Section 3/4.8.4 states that containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during normal plant operation (Modes 1 through 4) or by demonstrating the operability of primary and backup overcurrent protection circuit breakers during performance of periodic surveillances. It is proposed that Section 3/4.8.4.3 will be deleted since the containment electrical penetration and penetration conductors for these circuits are protected by primary and backup penetration circuit breakers which are demonstrated to be operable by periodic surveillance testing.

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

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 evaluated.

The revised Technical Specification Section 3.5.1 requirements will provide guidance to ensure that power to the accumulator isolation valves is removed when the accumulators are required to be operable and will clarify these requirements.

Removal of the electrical penetration protection requirements of Section 3/4.8.4.3 is justified since Section 3/4.8.4.1 (Containment Penetration Conductor Overcurrent Protective Devices) will provide guidance to ensure that two breakers in series protect the electrical penetrations and penetration conductors against an overcurrent condition and the single failure of a circuit breaker. The two breakers in series also protect the Class 1E buses against a variety of overcurrent conditions including electrical faults which may be introduced due to the possible submergence of the accumulator isolation valves during a LOCA [loss-of-coolant accident].

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

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

The amended Technical Specification Section 3.5.1 requirements will provide guidance to ensure that the accumulator isolation valves are deenergized when the accumulators are required to be operable. Deletion of the Technical Specifications Section 3.5.1 requires that electrical power to the safety injection accumulator isolation valves (3SIL*MV8808A, B, C, D) be removed for the accumulators to be operable. This requirement prevents the inadvertent closure of these isolation valves which would block the safety function of the accumulators. Section 4.5.1.c requires demonstrating accumulator operability by ''At least once per 31 days when the RCS [reactor coolant system] pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.'' The surveillance requirements for verifying removal of power to the accumulator isolation valves for Section 4.5.1.c will be changed to ''At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the associated circuit breakers are locked in a deenergized position or removed.''

The proposed change will clarify requirements for securing these breakers in the off (tripped) position in the applicable modes. In addition, index page xi has been revised to reflect the deletion of Section 3/4.8.4.3. Attachments 1 and 2 provide the mark-up and retyped pages of the Millstone Unit No. 3 Technical Specifications, respectively and reflect the currently issued version of the pages.

Millstone Unit No. 3 Technical Specifications Section 3/4.8.4.3 will not create the possibility of a new or different kind of accident from any accident previously evaluated since two breakers in series protect against an overcurrent condition and a single failure of a circuit breaker. The proposed amendment will not result in physical plant changes and there are no new credible failure modes. Therefore, the proposed amendment will 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 revised Technical Specification Section 3.5.1 will require that the accumulator isolation valves have their power deenergized when the accumulators are required to be operable. This requirement will maintain accumulator operability by assuring the accumulator isolation valves remain open.

The removal of the Millstone Unit No. 3 Technical Specification Section 3/4.8.4.3 is safe since redundant circuit breakers in series for the accumulator isolation valves will provide assurance that the electrical penetration and penetration conductors are protected against overcurrent conditions. This will provide assurance that the containment boundary is intact.

The proposed amendment will not adversely impact the physical protective boundaries (fuel matrix/cladding, RCS pressure boundary and containment) and therefore will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 9, 1995

Description of amendment request: The proposed amendment relocates Surveillance Requirement 4.6.6.1.d.3 for attaining a negative pressure in the secondary containment to Specification 3.6.6.2, Secondary Containment. The Action Statement of Section 3.6.6.1 is revised to decouple Sections 3.6.6.1 and 3.6.6.2. In addition, Definition 1.12, ''Secondary Containment Boundary'' is deleted and included in the Bases Section 3/4.6.6, Secondary Containment. Bases Section 3/4.6.6.2, Secondary Containment is expanded using the guidance of the improved standard technical specifications (STS) for Westinghouse plants (NUREG-1431).

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

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 evaluated.

The proposed changes to LCO [limiting condition for operation] 3.6.1.2, LCO 3.6.6.1 and LCO 3.6.6.2 Action Statements, relocation of Surveillance Requirement 4.6.6.1.d.3 to Specification 3.6.6.2, changes to Bases Section 3/4.6.6.1, 3/4.6.6.2, and 3/4.6.6.3, and deletion of Definition 1.12 will resolve the conflict that currently exists between Specifications 3.6.6.1 and 3.6.6.2. Specifically, the requirement to establish and maintain a negative pressure in the secondary containment boundary included in Specification 3.6.6.1 belongs to Specification 3.6.6.2. In the event Secondary Containment operability is not maintained, the Action Statement for LCO 3.6.6.2 requires that Secondary Containment operability must be restored within 24 hours. Twenty-four hours is a reasonable completion time considering the limited leakage design of containment and the low probability of a DBA [design basis accident] occurring during this time period. Therefore, it is considered that there exists no loss of safety function. The proposed changes do not modify the LCO or surveillance acceptance criterion, nor do they change the frequency of the surveillances. The proposed changes do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions. Therefore, the structures, systems, or components will perform their intended function when called upon. The proposed changes do not affect the probability of any previously evaluated accident. Additionally, the proposed changes are consistent with the new, improved STS for Westinghouse plants (NUREG-1431).

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

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

The proposed changes do not make any physical or operational changes to existing plant structures, systems, or components. The proposed changes do not introduce any new failure modes. The proposed changes simply resolve a conflict which currently exists between Specifications 3.6.6.1 and 3.6.6.2. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not have any adverse impact on the accident analyses. Also, the proposed changes resolve a conflict which currently exists between Specifications 3.6.6.1 and 3.6.6.2. The structures, systems, or components covered under Specifications 3.6.6.1 and 3.6.6.2 will performed [sic] their intended safety function when called upon.

Based on the above, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 20, 1995

Description of amendment request: The proposed amendment relocates the applicable requirements of Specification 3.6.3 for the main steam line isolation valves (MSIVs) to Specification 3.7.1.5, ''Main Steam Line Isolation Valves.'' In addition, the Applicability section of Specification 3.7.1.5 is revised to indicate that Specification 3.7.1.5 is applicable in Mode 1 and in Modes 2, 3 and 4, except where all MSIVs are closed and deactivated (i.e., in Modes 2, 3, and 4, Specification 3.7.1.5 is applicable only if the MSIVs are open). Also, the Action Statement for the Limiting Condition for Operation (LCO) 3.7.1.5 has been revised using the guidance of the improved standard technical specifications (STS) for Westinghouse plants (NUREG-1431).

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

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 evaluated.

The proposed changes to the Applicability section, Action Statements, and Surveillance Requirements of Specification 3.7.1.5 and the proposed changes to Specification 3.6.3 preserve the assumptions in the existing safety analysis. The proposed changes to the Applicability Section of Specification 3.7.1.5 will require the MSIVs to be operable in Mode 1 and in Modes 2, 3, and 4, except when closed and deactivated. The closure of the MSIVs in Modes 2, 3, or 4 is acceptable because when they are closed, they are already performing their safety function. Since the MSIV closure time has not been changed, there is no adverse impact on the accidents previously evaluated.

The proposed changes do not involve any physical changes to the plant, and do not alter the way any structure, system, or component functions. Therefore, the proposed changes do not affect the probability of any previously evaluated accident. Additionally, the proposed changes are consistent with the new, improved STS for Westinghouse plants (NUREG-1431).

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

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

The proposed changes do not make any physical changes to existing plant structures, systems, or components. When the MSIVs are closed and deactivated, they are already in the safe position; therefore, the proposed changes do not introduce a new failure mode. Additionally, the MSIV closure time (i.e., surveillance acceptance criterion) is not changed. The purpose of the surveillance is to ensure that the MSIVs can perform their safety function, and this requirement is preserved.

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

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

The proposed changes do not revise the closure time of the MSIVs. This provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of an accident. In addition, when they are closed in Modes 2, 3, and 4, they are already performing their safety function. Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 26, 1995

Description of amendment request: This proposed amendment would revise Technical Specification 2.3 to extend the allowed outage time (AOT) from 24 hours to 7 days for an inoperable low-pressure safety injection pump. This amendment request is a collaborative effort of participating Combustion Engineering Owners Group members and is based on an integrated assessment of plant operations and deterministic and probabilistic analyses.

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

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

The low pressure safety injection (LPSI) system is part of the emergency core cooling system. Inoperable LPSI components are not accident initiators in any accident previously evaluated. Therefore, these changes do not involve an increase in the probability of an accident previously evaluated.

The LPSI system is primarily designed to mitigate the consequences of a large loss of coolant accident (LOCA). These proposed changes do not affect any of the assumptions in the deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the LPSI allowed outage time (AOT) extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD-995, ''Combustion Engineering Owners Group Joint Applications Report for Low Pressure Safety Injection System AOT Extension.''

The CEOG report reviewed the risk factors that are impacted by extending the AOT for a single LPSI pump from 24 hours to seven (7) days, and demonstrates that the increase in risk is negligible. In order to perform a more complete assessment of the overall change in risk, an accounting for avoided risks associated with reducing power and going to hot or cold shutdown was also considered. This ''transition risk'' is important in understanding the trade-off between the risk of shutting down the plant compared with restoring a LPSI pump to operability while at power.

In assessing overall plant risk, the risk avoided based on LPSI system maintenance while in cold shutdown must also be considered. Every time the plant is placed in cold shutdown, the LPSI system is required for decay heat removal when in the shutdown cooling mode of operation. Maintenance performed on the LPSI system during shutdown cooling operations may add to the risk of a loss of shutdown cooling event. Therefore, performing LPSI system maintenance with the unit on-line, when the LPSI system is not normally in demand, represents a decrease in shutdown risk.

The CE study concluded that the change in core damage frequency due to increasing the LPSI AOT from 24 hours to seven (7) days is insignificant. Additionally, when the reduction in transition and shutdown risks are considered, it can be shown that there is an overall reduction in plant risk. Thus, it is the conclusion of the study that the overall plant impact will either be risk beneficial or risk neutral.

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

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

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. Therefore, the proposed changes do 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.

These proposed changes do not affect the limiting conditions for operation or their bases used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate this change. These evaluations demonstrate that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD-995. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: W. Dale Clark Library, 215

South 15th Street, Omaha, Nebraska 68102

Attorney for licensee: James R. Curtiss, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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

Date of amendment request: June 27, 1995

Description of amendment request: This proposed amendment would revise Technical Specification 2.2 on the chemical and volume control system to reformat, clarify the requirements, and be more consistent with Combustion Engineering Standard Technical Specifications (STS) as presented in NUREG-0212, Revision 2.

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

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

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes incorporate required actions, restrictions, and surveillance requirements for the Chemical and Volume Control System (CVCS) similar to Combustion Engineering Standard Technical Specifications (NUREG-0212 Revision 2).

Technical Specification (TS) 2.2(1) specifies the requirements for borated water sources and flow paths when the reactor is subcritical and fuel is in the reactor. In order for a flow path to be operable, a charging or high pressure safety injection pump is required to be operable to inject the boric acid solution into the Reactor Coolant System. Currently this specification does not state any operability requirements for boric acid transfer pumps, charging pumps or high pressure safety injection pumps. In addition, this specification does not state any required actions to be taken if the borated water source or flow path is not operable.

Therefore, the proposed changes incorporate requirements for the CVCS during shutdown into separate Limiting Conditions for Operations (LCOs) that will address the requirements for borated water sources, boric acid flow paths, charging pumps, and boric acid transfer pumps.

The proposed changes delete operability and surveillance requirements for level instrumentation on the boric acid storage tanks. Level instrumentation by itself does not fulfill a safety function. The proposed changes will still require verification of tank level.

Additionally, level instrumentation on the boric acid storage tanks does not meet any of the four criteria for inclusion into Technical Specifications as presented in the Final Policy Statement on Technical Specifications Improvements. This instrumentation is not installed instrumentation used to detect a significant degradation of the RCS boundary, a design feature or operating restriction that is an initial condition of a Design Basis Accident, a component that is part of the primary success path or actuates to mitigate a DBA, nor is it a component that has been shown to be significant to public health and safety. Therefore, testing and maintenance of the level instrumentation will be controlled outside of the TS.

TS 2.2(3) specifies the Modifications of Minimum Requirements that are allowed during Power Operation. This specification is inconsistent with TS 2.2(2) which states the minimum requirements and is incomplete as it does not address components during Modes 3, 4, and 5. The proposed changes incorporates consistent allowed outage times for the various components, and additional required actions for component inoperability during Modes 4 and 5 when fuel is in the reactor.

The proposed changes incorporate additional operability requirements for the CVCS and required actions to be taken for CVCS component inoperability during Modes 4 and 5 when fuel is in the reactor. The proposed changes delete inconsistencies and clarify operability requirements for the CVCS whenever the reactor coolant temperature (T_{cold}) is greater than or equal to 210 degrees F, and ensures that operation of the system is consistent with its design bases. The proposed changes also revise the allowed outage time for CVCS components from 24 hours to 72 hours based on Standard Technical Specifications. This change is insignificant based on the FCS plant specific probabilistic risk assessment. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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 this proposed change. No new modes of operation are proposed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes incorporate additional operability requirements, delete inconsistencies, and clarify operability requirements for the CVCS to ensure that operation of the system is consistent with its design bases. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: James R. Curtiss, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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

Date of amendment request: July 11, 1995

Description of amendment request: The proposed amendment would allow up to 24 hours to restore Safety Injection Tank (SIT) operability if the SIT is inoperable due to level and/or pressure outside prescribed limits or if the associated isolation valve is in other than the full open position. The proposed change would also allow up to 72 hours to restore SIT operability if the SIT is inoperable due to boron concentration outside prescribed limits.

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

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

The safety injection tanks (SITs) are passive components in the emergency core cooling system. The SITs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

SITs were designed to mitigate the consequences of a loss of coolant accident (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the affect of the SIT allowable outage time (AOT) extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD-994, "Combustion Engineering Owners Group Joint Applications Report for Safety Injection Tank AOT/STI Extension."

The AOT extension based upon boron concentration outside the prescribed limits does not involve a significant increase in the consequences of an accident as evaluated and approved by the NRC in NUREG-1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' This proposed change is applicable to FCS.

Therefore, the proposed changes would not increase 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.

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

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

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrated that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD-994. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

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NRC Project Director: William H. Bateman

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

Date of amendment request: June 23, 1995

Description of amendment request: This Technical Specifications (TS) Change Request involves a one-time (i.e., temporary) change affecting the Allowed Outage Time (AOT) for the Emergency Service Water (ESW) System; Residual Heat Removal Service Water (RHRSW) System; the Suppression Pool Cooling, the Suppression Pool Spray, and Low Pressure Coolant Injection (LPCI) modes of the Residual Heat Removal (RHR) System; and Core Spray System to be extended from 3 and 7 days to 14 days during the Limerick Generating Station (LGS), Unit 1, sixth refueling outage scheduled to begin January, 1996. This proposed extended AOT will allow adequate time to install isolation valves and cross-ties on the ESW and RHRSW Systems to facilitate future inspections or maintenance.

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

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

The proposed one-time TS changes will not increase the probability of an accident since it will only extend the time period that the 'A' ESW and RHRSW loops and the affected equipment can be out-of-service. The extension of the time duration that certain equipment is out-of-service has no direct physical impact on the plant. The proposed inoperable systems are normally in a standby mode while the unit is in OPCON 1 or 2 and are not directly supporting plant operation. Therefore, they can have no impact on the plant that would make an accident more likely to occur due to their inoperability.

During transients or events which require these systems to be operating, there is sufficient capacity in the operable loops to support plant operation or shutdown, in-so-much that failures that are accident initiators will not occur more frequently than previously postulated.

In addition, the consequences of an accident previously evaluated in the SAR [Safety Analysis Report] will not be increased. With the 'A' loops of ESW and RHRSW inoperable, a known quantity of equipment is either inoperable or the equipment is not fully capable of fulfilling its design function under all design conditions due to certain support systems not being operable. Based on the support functions of the ESW and RHRSW systems, a review of the plant was performed to determine the impacts that the inoperable ESW and RHRSW 'A' loops would have on other systems. The impacts were identified for each system, as discussed in the preceding Safety Assessment, and it was determined whether there were any adverse affects on the systems. It was then determined how the adverse affects would impact each system's design basis and overall plant safety. The consequences of any postulated accidents occurring on Unit 2 during this AOT extension was found to be bounded by the previous analyses as described in the SAR.

The existing AOTs limit the amount of time that the plant can operate with certain equipment inoperable, where single failure criteria is still met. The minimum equipment required to mitigate the consequences of an accident and/or safely shutdown the plant will be operable or the plant will be shutdown. Therefore, by extending certain AOTs and extending the assumptions concerning the combinations of events and single failures for the longer duration of each extended AOT, we conclude, based on the evaluations above, that at least the minimum equipment required to mitigate the consequences of an accident and/or safely shutdown the plant will still be operable during the extended AOT. Therefore, the consequences of an accident previously evaluated in the SAR will not be increased.

Therefore, these proposed one-time TS changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed one-time TS changes will not create the possibility of a different type of accident since it will only extend the time period that the 'A' ESW and RHRSW loops and the affected equipment can be out-of-service. The extension of the time duration that certain equipment is out-of-service has no direct physical impact on the plant and does not create any new accident initiators. The systems involved are either accident mitigation systems, safe shutdown systems or systems that support plant operation. All of the possible impacts that the inoperable equipment may have on its supported systems were previously analyzed in the SAR and are the basis for the present TS ACTION statements and AOTs. The impact of inoperable support systems for a given time duration was previously evaluated and any accident initiators created by the inoperable systems was evaluated. The lengthening of the time duration does not create any additional accident initiators for the plant.

Therefore, the proposed one-time TS changes will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The ESW and RHRSW systems and their supported systems are designed with sufficient independence and redundancy such that the removal from service of a component/subsystem will not prevent the systems from performing their required safety functions. Since removal of an ESW and a RHRSW loop from service with one unit in operation and the other unit in a refueling outage is allowed by the current Technical Specifications, then the concern is the reduced margin of safety incurred by extending the affected AOTs.

The present ESW and RHRSW AOT limits were set to ensure that sufficient safety-related equipment is available for response to all accident conditions and that sufficient decay heat removal capability is available for a LOCA/LOOP [Loss-of-Coolant Accident/Loss-of-Offsite Power] on one unit and simultaneous safe shutdown of the other unit. A slight reduction in the margin of safety is incurred during the proposed extended AOT due to the increased risk that an event could occur in a fourteen day period versus a three or seven day period. This increased risk is judged to be minimal due to the low probability of an event occurring during the extended AOT and based on the following discussion of minimum ECCS [Emergency Core Cooling System]/decay heat removal requirements.

The reduction in the margin of safety is not significant since the remaining operable ECCS equipment is adequate to mitigate the consequences of any accident. This conclusion is based on the information contained in the UFSAR [Updated Final SAR] reference documents NEDO-24708A and NEDC-30936-A. These documents describe the minimum requirements to successfully terminate a transient or LOCA initiating event (with scram), assuming multiple failures with realistic conditions were used to justify certain TS AOTs per UFSAR sections 6.3.1.1.2.o and 6.3.3.1. The minimum requirements for short term response to an accident would be either one LPCI pump or one Core Spray loop in conjunction with ADS [Automatic Depressurization System], which would be adequate to re-flood the vessel and maintain core cooling sufficient to preclude fuel damage. For long term response, the minimum requirements would be one loop of RHR for decay heat removal, along with another low pressure ECCS loop. These minimum requirements will be met since implementation of the proposed TS changes will require the operability of HPCI [High Pressure Coolant Injection], ADS, two LPCI subsystems (or one LPCI subsystem and one RHR subsystem during decay heat removal) and one Core Spray subsystem be maintained during the 14 day period. A Special Procedure will be written to ensure the operability of specified components and that other appropriate compensatory measures are implemented.

Compensatory measures will be taken prior to or during the proposed extended AOT for those fire regions that rely on one or more safe shutdown methods which would all be unable to safely shutdown the plant with inoperable loops of the ESW and RHRSW systems or the inoperable systems that ESW or RHRSW support. These compensatory measures will offset the increased risk of a fire event occurring in the vulnerable areas, during the fourteen day versus three day AOT period. Therefore, the proposed extended AOT does not adversely affect the approved level of fire protection as described in UFSAR Appendix 9A (Fire Protection Evaluation Report).

A Special Procedure will be written to administratively control the requirement to maintain the operability of specified components and implementation of any appropriate compensatory measures which are deemed necessary during the proposed AOT. In addition, operations personnel are fully qualified by normal periodic training to respond to and mitigate a Design Basis Accident, including the actions needed to ensure decay heat removal while LGS Unit 1 and Unit 2 are in the operational configurations described within this submittal. Accordingly, procedures are already in place that cover safe plant shutdown and decay heat removal for situations applicable to those in the proposed AOTs.

A Probabilistic Safety Assessment (PSA) Study was performed for an ESW and RHRSW loop being out-of-service for 14 days on an operating unit. The Core Damage Frequency (CDF) increased by $3.14x10^{-6}$, from $5.11x10^{-6}$ /reactor-year to $8.25x10^{-6}$/ reactor-year. In absolute terms, this is not a significant increase in risk. In addition, the modifications to be installed during this proposed extended AOT will allow for future maintenance and inspections to be performed on the ESW and RHRSW loops without removing an entire loop from service, which will reduce risk in the future. For example, if the ESW loop unavailability, due to testing or maintenance, is reduced by half, the CDF will decrease by more than four percent. It will also minimize the potential need for future AOT extensions on these systems.

Therefore, the implementation of the proposed one-time TS changes will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John F Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: September 29, 1994

Description of amendment request: The proposed Technical Specification changes represent revisions to Sections 3/4.3.7.2 ''Seismic Monitoring Instrumentation'' and 3/4.3.7.3 ''Meteorological Instrumentation.'' The proposed revisions remove the requirements from the Technical Specifications and relocates the appropriate descriptive information and testing requirements to the Hope Creek Updated Final Safety Analysis Report (UFSAR).

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

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

The proposed changes involve no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. Neither the relocation of the seismic/meteorological specifications to the UFSAR nor the elimination of the Special Report requirements represent changes that affect plant safety or alter existing accident analyses.

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

The proposed changes are procedural in nature concerning the operability and surveillance of instrumentation that are not safety related and will not impact the operation of any plant safety related component or equipment. Therefore, these changes will not create a new or unevaluated accident or operating condition.

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

In accordance with the guidance provided by the NRC regarding the improvement of Technical Specifications, SECY-93-067, the proposed changes relocate the seismic and meteorological instrumentation portions of the Technical Specifications, with the exception of the Special Report requirements, to the UFSAR. These instruments are not safety related and do not have any associated safety margins which could be affected by this change.

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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: November 23, 1994

Description of amendment request: The proposed changes to the Technical Specifications (TSs) would revise TS 4.8.2.1, ''Electrical Power Systems D. C. Sources, Surveillance Requirements,'' and associated Bases Section B 3/4.8.2. The proposed changes would (1) increase the terminal voltage acceptance criteria for the battery discharge test from 106 to 108 VDC, (2) delete a ''one time only'' test that is no longer applicable, (3) delete the battery load profile from the TS, and (4) revise TS Table 4.8.2.1-1, ''Battery Surveillance Requirements,'' to agree more closely with the BWR4 Standard Technical Specification format.

*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:

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

The proposed changes restore the conservatism to the battery voltage requirements by raising the minimum acceptable terminal voltage for the 125 VDC system in order to support proper operation of the connected loads. This change will cause no change in the probability of any accident and will, by providing increased support for connected loads, provide assurance [that] the consequences of previously evaluated accidents remain within limits. Removal of the load profile table does not affect the surveillance test loading which is contained in the station procedures. The (*) footnote deletion is purely editorial and has no safety bearing. Table changes agree with the format and wording of the improved BWR4 Standard Technical Specifications.

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

The revision of the battery sizing calculations did not change the design base requirement to supply the designed load for a duty cycle of 4-hours. The proposed change to the minimum acceptable battery terminal voltage for the 125 VDC system ensures proper voltages at the battery loads. No other changes to the physical plant or to the manner in which it is operated are caused by the proposed amendment; therefore, there is no new or different kind of accident created by this change.

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

The revision of the battery sizing calculations did not change the design base requirement to supply the designed load for a duty cycle of 4-hours; however, battery capacity sizing parameter of end cell voltage was changed to a more conservative value to account for minimum load voltage requirements. Load profiles for these batteries were slightly modified to incorporate more precise yet conservative load current values. These batteries were evaluated using a 25% additional capacity margin for aging as required by IEEE-450. In addition, the batteries have a design margin of 5 to 10% for load growth and/or less than optimum operating condition of the battery; thereby, maintaining safety margins. Additionally, changes are comparable to the format and ACTIONS of the improved BWR4 STS. Permitting 31 days to restore a battery to within CATEGORY A and/or B limits per the improved BWR4 STS does not involve a reduction in any margin of safety since the battery, in Category C, remains operable, as discussed in the BASES.

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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: November 28, 1994

Description of amendment request: The proposed Technical Specification (TS) revisions provide as follows: (1) The setpoints and allowable values for the Average Power Range Monitor (APRM) flow-biased upscale scram/control rod block would be modified to improve operating margin in the Extended Load Line Limit Analysis (ELLLA) region; (2) The proposed changes to the Rod Block Monitor (RBM) trip function would transfer control of the setpoint and allowable value for the RBM - upscale rod block to the Core Operating Limits Report (COLR); (3) For the Reactor Coolant System (RCS) recirculation flow upscale trip function, the proposed changes would revise the trip setpoint and allowable value to reflect 105% of rated core flow.

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

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

A. Changes to APRM Flow-Biased Scram/Control Rod Block

The proposed changes to the Average Power Range Monitor (APRM) flow-biased scram/control rod block setpoints and allowable values were evaluated using NRC approved procedures and methods. The results of this evaluation are demonstrated in NEDC-31487. Application of this change in APRM flow-biased scram/control rod block setpoints and allowable values to Reload 5/Cycle 6 is confirmed in General Electric Document No. 23A7219.

Analysis presented in NEDC-31487 demonstrate that performance in the ELLLA region is within design limits for overpressure protection, stability, loss-of-coolant, containment, reactor internals, flow-induced vibration, and reactor internal pressure difference. Impact of ELLLA operation on anticipated transients without scram is evaluated in Section 7.6.1.7.2 of the UFSAR. Application of ELLLA region extension to Reload 5/Cycle 6 has been confirmed in GE Document No. 23A7219.

Because operation with the APRM flow-biased scram/control rod block setpoints and allowable values is within the bases reviewed and approved by the NRC in the UFSAR [Updated Final Safety Analysis Report], this change does not significantly increase the possibility or consequences of an accident previously evaluated.

B. Transfer of RBM Setpoint Control to the COLR

The proposed changes would transfer control of the setpoint and allowable value for the rod block monitor (RBM) - Upscale rod block to the Core Operating Limits Report (COLR). Technical Specification 6.9.1.9, ''Core Operating Limits Report,'' requires that the analytical methods used to determine core operating limits be those previously reviewed and approved by the NRC and that the core operating limits be determined such that all applicable limits of the safety analysis are met.

The setpoint and allowable value incorporate a controlling value which will be specified in the COLR and noted as such by reference in the Technical Specifications. Therefore, the setpoint and allowable value would continue to be controlled in a manner that would ensure that safety analysis limits are met and implementation of the proposed changes would not reduce the level of assurance provided by the existing Technical Specifications. Based upon the above information, we conclude that implementation of the proposed change would not significantly increase the probability or consequences of an accident previously evaluated.

C. RCS Recirculation Flow Revisions

The original analysis used to support operation up to 105% of rated core flow is contained in NEDC-31487. NEDC-31487 addresses the full range of transient and accident events associated with operation up to 105% of rated core flow. The affects of operation with the revised RCS recirculation flow upscale trip setpoint and allowable value are bounded by the analysis presented in NEDC-31487.

In addition, cycle specific analysis performed for Reload 5/Cycle 6, have incorporated the assumption of operation up to 105% of rated core flow and have confirmed that operation is within allowable design limits.

Based on the above information, we conclude that the proposed change would not significantly increase the probability or consequences of an accident previously evaluated.

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

A. Changes to APRM Flow-Biased Scram/Control Rod Block

The proposed changes to the APRM flow-biased scram/control rod block setpoints and allowable values would not alter the function of the APRM system nor involve any type of plant modification. In addition, operation with the revised APRM flow-biased scram/control rod block setpoints and allowable values would not create any new operating modes, accident scenarios, equipment failure modes, or fission product release paths. Based upon the above information, we conclude that the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

B. Transfer of RBM Setpoint Control to the COLR

The proposed transfer of control of the RBM setpoint and allowable value to the COLR would not alter the function of the RBM system nor involve any type of plant modification. In addition, operation with the revised setpoint and allowable value would not create any new operating modes, accident scenarios, equipment failure modes, or fission product release paths. Based upon the above information, we conclude that the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. RCS Recirculation Flow Revisions

The proposed changes would not alter the function of the RCS recirculation flow upscale trip function nor involve any type of plant modification. In addition, operation with the revised RCS recirculation flow upscale trip setpoint and allowable value would not create any new operating modes, accident scenarios, equipment failure modes, or fission product release paths. Based upon the above information, we conclude that the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

A. Changes to APRM Flow-Biased Scram/Control Rod Block

The Bases for Hope Creek Technical Specification 2.2.1 state that the APRM setpoints were selected to provide adequate margin for the safety limits while allowing operating margins that reduce the possibility of unnecessary shutdowns.

The proposed changes would ensure that these objectives are met. The Minimum Critical Power Ratio (MCPR) operating limit specified in the Hope Creek COLR was determined using the APRM flow-biased scram/control rod block setpoints and allowable values proposed in this amendment application and has been chosen to ensure that the cladding safety limit would not be violated during normal plant operations and anticipated transients. Since the operating limit MCPR is chosen such that the cladding safety limit is maintained, adequate margins for the safety limits are ensured. The proposed changes would also serve to ensure that the objective of avoiding unnecessary shutdowns is met by furnishing greater margin between the operating envelope and the setpoint at lower flows.

Based on the above information, we conclude that the proposed changes would not significantly reduce a margin of safety.

B. Transfer of RBM Setpoint Control to the COLR

The proposed transfer of control of the RBM setpoint and allowable value to the COLR would not affect the methodology for establishing the core operating limits. The setpoint and allowable value are modified to incorporate a controlling value which will be included in the COLR and indicated as such by reference in the Technical Specifications. Therefore, the setpoint and allowable value would continue to be controlled in a manner that would ensure that safety analysis limits are met. We conclude that implementation of the proposed changes would not significantly reduce a margin of safety.

C. RCS Recirculation Flow Revisions

The HCGS was licensed to operate up to 105% of rated core flow as part of Amendment 15. The analysis used to justify operation up to 105% of rated core flow is contained in NEDC-31487. NEDC-31487 addresses the full range of transient and accident events associated with operation up to 105% of rated core flow. The affects of operation with the revised RCS recirculation flow upscale trip setpoint and allowable value are bounded by the analysis presented in NEDC-31487.

In addition, cycle specific analysis performed for Reload 5/Cycle 6, have incorporated the assumptions of operation up to 105% of rated core flow and have confirmed that operation is within allowable design limits.

Based on the above information, we conclude that the proposed changes would not significantly reduce 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: Pennsville Public Library,190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: January 11, 1995

Description of amendment request: The proposed Technical Specification (TS) revision provides changes to TS Section 3/4.3.8 ''Turbine Overspeed Protection System.'' The proposed revision removes these requirements from the TS and relocates the Bases to the Hope Creek Updated Final Safety Analysis Report (UFSAR) and the Surveillance Requirements to the applicable surveillance procedures. The Limiting Conditions for Operation (LCOs) would be eliminated.

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

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

The proposed changes involve no hardware changes, no changes to existing structures, and no changes to the operation of any systems or components. Specifically, the deletion of the LCO's by this submittal will not alter established turbine overspeed protection system operation. Procedural guidance will be provided in the event of an inoperable control, stop, or intermediate valve to place the system in a safe condition. The relocation of this specification to the UFSAR and surveillance procedures will continue to ensure that the probability of unacceptable damage to safety-related structures, systems, and components from turbine missiles remains acceptably low. Relocation of this specification's Bases and Surveillance Requirements to the UFSAR and surveillance procedures, respectively, and the deletion of the LCO's represents changes that do not affect plant safety and do not alter existing accident analyses.

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

The proposed changes are procedural in nature concerning the location of the descriptive information and surveillance requirements for the turbine overspeed protection system. Removing these specifications from the Technical Specifications and placing them in the UFSAR and surveillance procedures will not alter the operation of the turbine overspeed protection system or its ability to perform its intended function. Procedural guidance will be provided to assist in placing the system in a safe condition while maintenance and testing of this system will continue in accordance with the turbine manufacturers recommendations taking into consideration plant operating experience and ASME guidance. Therefore, these changes will not create a new or unevaluated accident or operating condition.

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

The proposed changes relocate the Turbine Overspeed Protection System portion of the Technical Specifications to the UFSAR and surveillance procedures in accordance with guidance provided by the NRC Final Policy Statement regarding the improvement of Technical Specifications. The requirements that will reside in the UFSAR for the turbine overspeed protection system will ensure that the system remains capable of protecting against excessive turbine overspeed. Therefore, the proposed changes will not involve a significant reduction in any 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: January 20, 1995

Description of amendment request: The proposed Technical Specification (TS) revision represents changes to TS Section 3/4.11.2.6 ‘‘Explosive Gas Mixture,’’ TS Table 3.3.7.11-1 ‘‘Radioactive Gaseous Effluent Monitoring Instrumentation,’’ and TS Table 4.3.7.11-1 ‘‘Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements.’’ The proposed revision would remove these TS from the Technical Specifications and relocate the Bases to the Hope Creek Updated Final Safety Analysis Report (UFSAR) and the Surveillance Requirements to the applicable surveillance procedures. The Limiting Conditions for Operation (LCOs) would be eliminated.

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

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

The proposed changes involve no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. The relocation of this specification to the UFSAR and surveillance procedures will continue to ensure that the entrainment of hydrogen in the main condenser is monitored and controlled. Relocation of this specification’s Bases and Surveillance Requirements to the UFSAR and surveillance procedures, respectively, and the deletion of the LCO’s represent changes that do not affect plant safety and do not alter existing accident analyses.

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

The proposed changes are procedural in nature concerning the location of the descriptive information and surveillance requirements for the explosive gas mixture monitoring instrumentation. Removing these specifications from the Technical Specifications and placing them in the UFSAR and surveillance procedures will not alter the operation of the explosive gas monitors or their ability to perform intended functions. Maintenance and testing of these monitors will continue based upon the manufacturers’ recommendations taking into consideration plant operating experience. Therefore, these changes will not create a new or unevaluated accident or operating condition.

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

The proposed changes relocate the Explosive Gas Mixture specifications from the Technical Specifications to the UFSAR and surveillance procedures in accordance with guidance provided by the NRC Final Policy Statement regarding the improvement of Technical Specifications. The requirements that will reside in the UFSAR and surveillance procedures for the explosive gas mixture monitoring instrumentation will ensure that the ability to determine main condenser hydrogen concentrations is properly maintained. Therefore, the proposed changes will not involve a significant reduction in any 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: January 20, 1995

Description of amendment request: The proposed change to the Technical Specifications (TS) would revise TS 4.1.3.1.2.b, ‘‘Control Rods - Surveillance Requirement’’ to change the required action to be taken when a control rod becomes immovable due to excessive friction or mechanical interference from ‘‘at least once per’’ 24 hours to ‘‘within’’ 24 hours. The other control rods would be tested within 24 hours and every 7 days thereafter, as opposed to the current requirement of testing every 24 hours.

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

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

The proposed change involves no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. The revision of the control rod movement test frequency represents a change that does not affect plant safety and does not alter existing accident analyses.

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

The proposed change is procedural in nature concerning the frequency of control rod movement tests for all withdrawn control rods after a control rod has been determined to be immovable due to excessive friction or mechanical interference. The methodology for determining additional immovable control rods remain unchanged. The proposed change while slightly increasing the possibility of an undetected immovable control rod will not create a new or unevaluated accident or operating condition.

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

The proposed change is in accordance with recommendations provided by the NRC regarding the improvement of Technical Specifications. This change will result in the perpetuation of current safety margins while reducing regulatory burden and decreasing equipment degradation.

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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: June 1, 1995

Description of amendment request: The proposed change would revise the Technical Specifications to make them more restrictive regarding control rod drive (CRD) scram time testing. CRD scram time testing would be required following maintenance prior to considering the CRD operable, and could be performed at any reactor pressure. Additional testing would be required when reactor coolant pressure is greater than or equal to 950 psig and prior to 40 percent rated thermal power.

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

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

The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analysis and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more restrictive requirements either has no impact on or increase in the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition providing additional restrictions to enhance plant safety. The change maintains requirements within safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: April 10, 1995

Description of amendment request: The proposed amendment would revise the following Technical Specifications (TS) and their associated Bases: TS 3/4.7.1.2, ''Auxiliary Feedwater System,'' to clarify Action ''a'' by inserting ''or both'' steam generator≥s'' and to remove references to pressure indicators and specific pressure readings and adding performance based requirements; TS 3/4.7.1.3, ''Condensate Storage Tanks,'' to modify the Limiting Condition for Operation (LCO) to more closely conform to standard TS; and TS 3/4.7.1.7, ''Motor Driven Feedwater Pump System,'' to consolidate the requirements of 2 current surveillance requirements and clarify the operability requirements when local manual valves are realigned for testing purposes.

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

Toledo Edison has reviewed the proposed changes and determined that asignificant hazards consideration does not exist because operation of the Davis-besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previous analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed. The proposed changes are clarifications and the incorporations of the guidance provided by NUREG-1430. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated and, no new or different failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed changes. No new or different types of failures or accident initiators are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed changes are clarifications and the incorporations of the guidance provided by NUREG-1430, and continue to ensure the availability and capability of the Auxiliary Feedwater System, Service Water System and the Motor Driven Feedwater Pump System when called upon to perform their functions. The proposed changes will not adversely impact any safety analysis assumptions.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 1, 1995

Description of amendment request: The proposed amendment would change the allowed outage time from 72 hours to 7 days for one unavailable emergency diesel generator (EDG) as detailed in Technical Specification 3.8.1.1, ''AC Power Sources, Operating,'' and its associated Bases 3.0.5.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed change to increase the allowed outage time for one emergency diesel generator from three (3) days to seven (7) days does not make a change to any accident initiator, initiating condition or assumption. The accident previously evaluated in the DBNPS Updated Safety Analysis Report (USAR) Section 15.2.9, Loss of All AC Power to the Station

Auxiliaries (Station Blackout), is not affected by this proposed change. The proposed change does not involve a significant change to the plant design or operation, only to the allowed outage time, and based on a review of the available alternate A.C. power sources, the effect on probabilistic risk at power, the effect on shutdown risk, and maintenance planning and scheduling, this change has been determined to be acceptable.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not invalidate assumptions used in evaluating the radiological consequences of an accident, does not alter the source term or containment isolation and does not provide a new radiation release path or alter potential radiological releases.

2. Not create the possibility of a new or different kind of accident from any previously evaluated because the proposed change does not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in the margin of safety because the proposed change does not significantly reduce the margin to safety which exists in the present Technical Specification action statements. The DBNPS USAR Section 15.2.9 evaluates the acceptability of the loss of all A.C. power to the station, including the loss of both EDGs, and the margin of safety in this analysis is not affected by the proposed change. in addition, since the issuance of the original DBNPS Operating License Technical Specifications Toledo Edison has installed a Station Blackout Diesel Generator (SBODG), comparable in continuous rating to the EDGs and capable of providing emergency A.C. power in the event all three offsite 345 kV transmission lines and the two EDGs are unavailable. This has positive effect on maintaining the margin to safety which exists in the Technical Specifications with a three day allowed outage time, which was established prior to installation of the SBODG.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 7, 1995

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.9.4, Refueling Operations - Containment Penetrations, and associated Bases 3/4.9.4, Containment Penetrations. The proposed changes include revising the Limiting Condition for Operation (LCO) 3.9.4.b to allow both doors of the containment personnel airlock to be open during core alterations or movement of irradiated fuel within the containment, provided that certain specified conditions are meet. Additional changes are proposed to revise or clarify TS LCO 3.9.4.c, TS Action 3.9.4.a, and TS Surveillance Requirement 4.9.4, and modify the Bases to reflect the requested changes.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no Updated Safety Analysis Report (USAR) accident initiators are affected by the proposed changes.

The proposed change to TS LCO 3.9.4.b would allow both doors of the containment personnel air lock to be open during core alterations or movement of irradiated fuel within the containment, provided that certain specified conditions are met. The containment personnel air lock is not an initiator to any accident. Whether the containment personnel air lock doors are open or closed during fuel movement and core alterations has no effect on the probability of any accident previously evaluated.

The proposed clarification of TS LCO 3.9.4.c, changing the term ''outside atmosphere'' to ''atmosphere outside containment,'' and the proposed change to TS LCO 3.9.4.c.1, confirming that, in addition to a manual or automatic isolation valve, or a blind flange, equivalent means may be used to close a containment penetration, have no bearing on the probability of an accident previously evaluated.

The proposed changes to TS Action 3.9.4.a, TS Surveillance Requirement (SR) 4.9.4, and TS Bases 3/4.9.4 are administrative changes and have no bearing on the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate accident conditions or assumptions used in evaluating the radiological consequences of any accident.

he analysis results for a fuel handling accident inside containment, as presented in Section 15.4.7.3 of the DBNPS USAR, are well within the 10 CFR 100 guideline values. Since the analysis does not take credit for containment isolation, the status of the personnel air lock has no impact on the acceptability of the results. In the event of a fuel handling accident, release of radioactive material will continue to be minimized since the air lock door will remain capable of being closed. Further, the proposed change could significantly reduce the dose to workers in the containment in the event of a fuel handling accident by speeding the containment evacuation process.

Since an engineering evaluation described in proposed Bases 3/4.9.4 will ensure that a particular containment penetration closure technique is capable of restricting the release of radioactive material from a fuel handling accident, the proposed change to TS LCO 3.9.4.c.1, confirming that an equivalent means may be used to close a containment penetration, has no adverse effect on the consequences of an accident previously evaluated.

The proposed clarification of TS LCO 3.9.4.c, and the proposed changes to TS Action 3.9.4.a, TS SR 4.9.4, and TS Bases 3/4,9.4 are administrative changes and have no effect on the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no new failure modes or mechanisms associated with the proposed changes, nor do the proposed changes involve any modification of plant equipment or changes in plant operational limits.

As described above, the analysis results for a fuel handling accident inside containment does not take credit for containment isolation. Thus the proposed change to TS LCO 3.9.4.b to allow both doors of the containment personnel air lock to be open during core alterations or movement of irradiated fuel within the containment could affect the release path for radioactive material released during a fuel handling accident, however no new or different kind of accident will result.

3. Not involve a significant reduction in the margin of safety.

The analysis results for a fuel handling accident inside containment, as presented in [Section 15.4.7.3 of] the DBNPS USAR, are well within the 10 CFR 100 guideline values. Since the analysis does not take credit for containment isolation, the status of the personnel air lock has no impact on the acceptability of the results.

The proposed change to TS LCO 3.9.4.c.1 regarding the use of equivalent means of containment penetration closure has no adverse impact on the margin of safety since an equivalent containment penetration closure technique will provide the same assurance of containment closure during core alterations or movement of irradiated fuel inside containment.

The various administrative changes and clarifications proposed will not reduce the margin of safety.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 23, 1995

Description of amendment request: The proposed amendment would relocate Technical Specifications (TS) 3/4.3.3.3 - Seismic Instrumentation, TS 3/4.3.3.4 - Meteorological Instrumentation, and TS 3/4.4.11 - Reactor Coolant System Vents and associated Bases.

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit Number 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previous analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

The proposed changes are deletions and relocations of specifications that do not meet the NRC Final Policy Statement [58 FR 39132, dated July 22, 1993] criteria for inclusion in TS. Furthermore, these relocations and deletions are consistent with the NRC guidance for TS provided by the ''Improved Standard Technical Specifications for Babcock and Wilcox Plants,'' NUREG-1430, Revision 0. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed changes. No new or different types of failures or accident initiators are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because Seismic Instrumentation, Meteorological Instrumentation, and Reactor Coolant System Vents are not inputs in the calculation of any safety margin with regard to TS Safety Limits, Limiting Safety System Settings, other TS Limiting Conditions for Operation, or other previously defined margins for any structure, system, or component important to safety.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: July 14, 1995

Description of amendment request: The proposed Technical Specifications (TS) changes would provide a two-hour allowed outage time (AOT) for one residual heat removal (RHR) pump to accommodate plant safety and emergency power systems surveillance testing and permit depressurizing safety injection (SI) accumulators in lieu of accumulator isolation.

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

Specifically, operation of the Surry Power Station in accordance with the proposed change will not:

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

Surveillance and testing requirements are necessary to assure that RHR and interfacing systems' reliability is maintained. Existing analyses demonstrate that adequate shutdown cooling will be maintained with one train of RHR Operable and in service. Analyses also demonstrate that alternate shutdown cooling modes remain available with adequate decay heat removal capability. Furthermore, the opposite train of RHR remains available while in the two hour surveillance AOT. The response time and operator actions required to place the available RHR train in service are consistent with similar operator response times and actions

required to place alternate shutdown cooling modes in service. The administrative controls and procedures in place assure adequate shutdown cooling capability is maintained as supported by existing analyses.

The existing safety analyses demonstrate that Reactor Coolant System [RCS] integrity will be maintained when SI accumulator pressure is below the pressurizer PORV [power operated relief valve] LTOPS [low temperature overpressure system] setpoint. Therefore, SI accumulator isolation is not required to ensure Reactor Coolant System integrity. With RCS temperature below the LTOPS enabling temperature, automatic actuation of the pressurizer PORVs or other TS specified relief paths ensure the assumed design basis reactor vessel beltline flaw will not propagate under design basis low temperature overpressurization accident conditions. System design and configuration adequately mitigate an LTOPS actuation due to an SI accumulator discharge with no negative consequences regarding RCS structural integrity or SBLOCA [small break loss-of-coolant accidents] concerns.

Therefore, the proposed Allowed Outage Time for an inoperable RHR loop and the ability to depressurize the SI accumulator in lieu of SI accumulator isolation do not increase the probability or consequence of any previously analyzed accidents.

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

The proposed two hour AOT for one train of the RHR System will preclude the possibility of a Technical Specification violation for conditions where a train of RHR is out of service for surveillance testing. Calculations by Westinghouse with evaluations and supporting analyses performed by Virginia Power, confirm the adequacy of decay heat removal with one RHR train in service, and multiple alternate shutdown cooling modes remain available. There are no plant modifications required by this proposed TS change. Further, the proposed change does not invalidate any component design criteria or the assumptions of the UFSAR [updated final safety analysis report] accident analyses. The RHR System is being operated in a manner consistent with the design basis and configuration of the system and is supported by existing analyses and procedural controls.

There are no new failure modes or mechanisms associated with the proposed change to allow the depressurizing of a SI accumulator to a pressure value below the LTOPS setpoint. The LTOPS enabling temperature remains unchanged. No operating limits or setpoints are added or deleted by the proposed change. Reactor Coolant System pressure relief paths are not affected.

Therefore, the possibility of a new or different kind of accident is not being created by the proposed Allowed Outage Time for an inoperable RHR loop and the ability to depressurize the SI accumulator in lieu of SI accumulator isolation.

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

The proposed Technical Specifications change does not involve a reduction in a margin of safety. The existing safety analyses demonstrate that adequate shutdown cooling will be maintained when a train of RHR is out of service for up to two hours for plant system surveillance testing, while the operable train of RHR is operating. Supporting analyses determined that the RHR System meets the design cooldown requirements for a reactor core rating of 2546 MWth [megawatt thermal] with either one or both trains of RHR in service. Additionally, an evaluation of the technical basis for shutdown operations for the proposed Surry core uprating to 2546 MWth determined that the administrative controls and Abnormal Procedures in place at Surry ensure adequate decay heat removal capability during shutdown conditions. The administrative controls and procedure revisions are supported by a detailed series of thermal-hydraulic calculations for various loss of RHR scenarios. There is no reduction in shutdown cooling capability due to the proposed TS change, and no reduction in the capability to mitigate a loss of decay heat removal event since the RHR train affected by the testing is available and can be restored in a comparable time period to that required to restore RHR to service in the event of loss of station power or loss of the operating train of RHR. Consequently, system design, plant configuration, and administrative controls remain available to adequately mitigate a loss of RHR event with a single train of RHR out of service for up to two hours during plant system surveillance testing. It may be concluded that there is no reduction in the margin of safety due to the proposed Technical Specification change.

Existing safety analyses also demonstrate that Reactor Coolant system integrity will be maintained in the event of an inadvertent SI accumulator discharge when SI accumulator pressure is below the pressurizer PORV LTOPS setpoint. Sufficient administrative controls are maintained to ensure LTOPS is ''Enabled'' and SI accumulators are isolated at the appropriate RCS conditions to minimize the possibility of challenging RCS integrity. Technical Specifications administrative controls that prevent inadvertent charging pump operation, maintain adequate relief paths, and restrict Steam Generator primary to secondary temperature differential remain in place. Consequently, the Technical Specifications change ensures that an inadvertent SI accumulator discharge cannot challenge RCS structural integrity during LTOPS conditions when SI accumulator pressure is below the pressurizer PORV LTOPS setpoint.

Therefore, the proposed Allowed Outage Time for an inoperable RHR loop and the ability to depressurize the SI accumulator in lieu of SI accumulator isolation does not reduce any margin of safety as defined in the Technical Specifications.

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: June 14, 1995, as supplemented by letter dated July 13, 1995.

Description of amendment request: This amendment request proposes to revise Technical Specification (TS) 3.2.3, ''Nuclear Enthalpy Rise Hot Channel Factor,'' TS 6.9.1.9, ''Core Operating Limits Report,'' and the associated Bases sections. The revisions are needed to incorporate changes associated with the planned implementation of advanced nuclear and core thermal-hydraulic design methodologies licensed from Westinghouse Electric Corporation for core reload design, starting with Cycle 9.

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

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

The probability of occurrence and the consequences of an accident evaluated previously in the Updated Safety Analysis Report (USAR) are not increased due to the proposed technical specification changes. The Technical Specification changes being requested are to reflect revised calculational methods to be used for core reload design, starting with Cycle 9. There are no changes being made to any licensed design parameters from previous cycles. Thus, it is concluded that the probability and consequences of the accidents previously evaluated in the USAR are not increased.

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

There is no new type of accident or malfunction being created. The proposed changes only provide revised analysis methodologies to support core reload design, starting with Cycle 9. The requested changes do not change the method and manner of plant operation. The safety design bases in the USAR have not been altered. Thus, the requested changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not change the plant configuration in a way that introduces a new potential hazard to the plant and do not involve a significant reduction in the margin of safety. The analyses and evaluations discussed in the safety evaluation (Attachment I) [Attached to Wolf Creek Nuclear Operating Corporation's letter number ET 95-0051, dated June 14, 1995] demonstrates that all applicable design criteria continue to be met for the changes. Therefore, it is concluded that the margin of safety, as described in the bases to any technical specification, 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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

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

NRC Project Director: William H. Bateman

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.

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

Date of amendment request: February 23, 1995

Description of amendment request: The amendment relates to Commonwealth Edison Company's (ComEd) request to reflect the merger between IIGEC, MidAmerican, Midwest Power Systems Inc., and Midwest Resources, Inc. By letter dated November 21, 1994, Iowa-Illinois Gas and Electric Company (IIGEC) requested approval, pursuant to Section 50.80 of Title 10 of the Code of Federal Regulations, of the transfer of its ownership share of 25 percent of Quad Cities Nuclear Power Station, Units 1 and 2, to MidAmerican Energy Company (MidAmerican).

Date of publication of individual notice in **Federal Register:** July 5, 1995 (60 FR 35054)

Expiration date of individual notice: August 4, 1995

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

Date of amendment request: March 31, 1995

Description of amendment request: The proposed amendment will delete Technical Specification (TS) Sections 1.38 and 1.39, ''Definitions, Fuel Assembly Types,'' revise TS Sections 3/4.9.3, ''Refueling Operations, Decay Time'' and TS 3/4.9.14, ''Refueling Operations, Spent Fuel Pool - Reactivity Condition,'' replace TS Sections 5.6.1.1, ''Spent Fuel,'' and TS 5.6.3, ''Capacity,'' and add a new TS Section 3/4.9.15, ''Refueling Operations, Spent Fuel Pool Cooling.'' These changes would support a rerack of the spent fuel pool to expand the spent fuel pool's storage capacity from 1168 assemblies to 1480 assemblies so as to accommodate a full-core-discharge through the current validity date of the Haddam Neck operating license (2007).

Date of publication of individual notice in **Federal Register:** May 12, 1995 (60 FR 25746)

Expiration date of individual notice: June 12, 1995

Local Public Document Room location: Russell Library, 123 Broad Street, Middletown, CT 06457.

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: June 14, 1995

Brief description of amendments: The amendments revise the requirement to perform an emergency diesel generator (EDG) automatic start and sequence loading test immediately following the 24 hour EDG endurance test.

Date of issuance: July 18, 1995

Effective date: July 18, 1995

Amendment Nos.: 166 and 154

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration determination: Yes (60 FR 34308). This notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by July 31, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendments, finding of exigent circumstances and final no significant hazards consideration determination is contained in a Safety Evaluation dated July 18, 1995.

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

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

Date of application for amendments: March 31, 1995

Brief description of amendments: The amendments revise Technical Specification section 3.9.4 to allow, under certain conditions, both containment personnel airlocks to be open during core alterations.

Date of issuance: July 12, 1995

Effective date: July 12, 1995

Amendment Nos.: 197 and 182

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29879)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 12, 1995.No significant hazards consideration comments received: No.

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

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station,Nemaha County, Nebraska

Date of amendment request: May 2, 1995

Brief description of amendment: The amendment revised Surveillance Requirement 4.7.A.2.f.1 to allow a one-time extension for the performance of Type B local leak rate testing of the drywell head and manport from July 17, 1995, until startup from Refueling Outage 16, scheduled to commence on October 13, 1995.

Date of issuance: July 11, 1995

Effective date: July 11, 1995

Amendment No.: 170

Facility Operating License No. DPR-46. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29879)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 1995No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, NE 68305.

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

Date of application for amendment: January 10, 1995

Brief description of amendment: The amendment revises the Technical Specifications to delete the power range negative flux trip from Tables 2.2-1, 3.3-1, and 4.3-1, and delete the associated Bases Section 2.0.

Date of issuance: July 11, 1995

Effective date: As of the date of issuance to be implemented within30 days.

Amendment No.: 116

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

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11135)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 29, 1995

Brief description of amendment: The amendment revises Technical Specification 3.10.5 to allow more than one control bank to be fully withdrawn from the core simultaneously in order to conduct rod drop time response testing.

Date of issuance: July 11, 1995

Effective date: As of the date of issuance to be implemented within60 days.

Amendment No.: 117

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

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29880) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 12, 1993, as supplemented by letters dated March 22, 1993, and August 25, 1994

Brief description of amendment: The amendment increases the minimum core spray pump flow to more conservatively account for emergency core cooling systems bypass leakage paths. The amendment also makes various typographical, editorial and administrative corrections and changes.

Date of issuance: July 12, 1995

Effective date: July 12, 1995

Amendment No.: 93

Facility Operating License No. DPR-22. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41508). The August 25, 1994 letter provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards considerations determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 1995.No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 11, 1994, as supplemented by letters dated April 7, 1995, and June 26, 1995

Brief description of amendment: The amendment implements administrative changes to TS 5.2 and 5.5. These changes reflect organizational changes in OPPD senior management, delete specific titles of personnel on the Plant Review Committee (PRC), revise the makeup of the PRC quorum, revise the membership of the Senior Audit and Review Committee (SARC), delete SARC audit frequencies and add minor clarifications to the descriptions of SARC reviews and audits.

Date of issuance: July 21, 1995

Effective date: July 21, 1995

Amendment No.: 168

Facility Operating License No. DPR-40. Amendment revised the TechnicalSpecifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65819). The April 7, 1995, and June 26, 1995, letters provided clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 1995.No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 1, 1995

Brief description of amendment: The amendment revises TS 2.5, 2.8, 2.11, 3.2, and 3.10 and relocates administrative controls for the emergency and security plans from TS 5.5 and 5.8 to the plans. The relocation is in accordance with Generic Letter (GL) 93-07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans.''

Date of issuance: July 21, 1995

Effective date: July 21, 1995

Amendment No.: 169

Facility Operating License No. DPR-40: The amendment revised the TechnicalSpecifications.

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18627) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 10, 1995

Brief description of amendment: This amendment revised License No. DPR-7, to permit the provisions of 10 CFR 50.59 to be applied with respect to changes to the facility or procedures described in the Decommissioning Plan or changes to the Decommissioning Plan, and the conduct of tests or experiments not described in the Decommissioning Plan.

Date of issuance: July 7, 1995

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

Amendment No.: 29Facility License No. DPR-7: This amendment revised License No. DPR-7

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29885)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 7, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Humboldt County Library, 636 F Street, Eureka, California 95501.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: November 21, 1994

Brief description of amendment: This amendment changes the technical specifications (TS) by allowing the third Type A Containment Integrated Leakage Rate Test in the second 10-year service period to be conducted during refueling outage 11 scheduled for September 1997. This TS change is consistent with a one-time exemption from Appendix J to 10 CFR Part 50 that extends the 10-year service period and allows the three type A tests to be performed at intervals that are not approximately equal.

Date of issuance: July 10, 1995

Effective date: July 10, 1995

Amendment No.: 210

Facility Operating License No. DPR-56: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 23, 1995 (60 FR 27340)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 1995.No significant hazards consideration comments received: No

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

PECO Energy Company, Public Service Electric and Gas Company,Delmarva Power and Light Company, and Atlantic City Electric Company,Docket No. 50-278, Peach Bottom Atomic Power Station,Unit No. 3, York County, Pennsylvania

Date of application for amendment: June 23, 1993, as supplemented by letters dated April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28 (two letters), September 16, September 30, and October 14, 1994 and June 22, 1995.

Brief description of amendment: The amendment raises the authorized maximum power level from 3293 MWt to a new limit of 3458 MWt. The amendment also approves changes to the Technical Specifications to implement operation at the increased power limit.

Date of issuance: July 18, 1995

Effective date: As of date of issuance and is to be implemented prior to startup in Cycle 11, currently scheduled for October 1995.

Amendment No.: 211

Facility Operating License No. DPR-56: Amendment revised the License and the Technical Specifications.

Date of initial notice in **Federal Register:** August 29, 1994 (59 FR 44432)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

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: April 10, 1995

Brief description of amendments: Remove the response time limit Tables 3.3.1-2, 3.3.2-3, and 3.3.3-3 from the Technical Specifications, and add the information to the Final Safety Analysis Report in accordance with Generic Letter 93-08.

Date of issuance: July 11, 1995

Effective date: July 11, 1995

Amendment Nos.: 148 and 118

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

Date of initial notice in **Federal Register:** June 6, 1995 (60FR 29887)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 11, 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.

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

Date of application for amendments: November 21, 1994, as supplemented April 6, and July 3, 1995

Brief description of amendments: These amendments make changes affecting the Administrative Controls Section of the Technical Specifications. The areas changed are Nuclear Effectiveness and Efficiency Design Study (NEEDS) Organization Title Changes; Minimum Shift Crew Composition; delete Independent Technical Review Section from TS; delete Nuclear Review Board (NRB) Review Section from TS; and delete NRB Audit Section from TS.

Date of issuance: July 18, 1995

Effective date: Units 1 and 2, as of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 96 and 60

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24914)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 31, 1994, as supplemented July 3, 1995

Brief description of amendments: These amendments modify TS Sections 3.4.9.1, 3.4.9.2, 3.9.11.1, 3.9.11.2, and the associated Bases Sections 3/4.4.9 and 3/4.4.11, to permit the use of either an ''analytical approach'' (i.e., calculation) or ''demonstrations'' to ensure the operability of an alternate decay heat removal method, rather than the existing TS requirement which stipulates that operability of the alternate decay removal method be demonstrated.

Date of issuance: July 18, 1995

Effective date: Units 1 and 2, as of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 97 and 61

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 22, 1994, as supplemented July 3, 1995

Brief description of amendments: These amendments revise Technical Specification Surveillance Requirement 4.1.3.1.4a to delete the requirement that the Scram Discharge Volume (SDV) be determined operable by testing the SDV vent and drain valves from a configuration of less than or equal to 50% rod density.

Date of issuance: July 18, 1995

Effective date: Units 1 and 2, effective as of date of issuance and shall be implemented within 30 days.

Amendment Nos.: 98 and 62

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55881)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 22, 1994, as supplemented by letter dated July 3, 1995

Brief description of amendments: The amendments revise the Technical Specifications surveillance requirements for scram insertion times and revise the TS surveillance requirements for control rod block and source range monitoring instrumentation.

Date of issuance: July 18, 1995

Effective date: Units 1 and 2, effective as of the date of issuance and shall be implemented within 30 days.

Amendment Nos.: 99 and 63

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55881)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station (FSV), Unit No. 1, Platteville, Colorado

Date of application for amendment: Amendment No. 88, April 14, 1995.

Brief description of amendment: This amendment would revise the FSV Decommissioning Technical Specifications (DTS) by: revising the FSV DTS to reflect recent organizational changes resulting from corporate restructuring to prepare for repowering the site with natural gas-power turbines and to incorporate editorial changes. The staff has determined that the proposed amendment does not require a significant hazard consideration, pursuant to 10 CFR 50.92.Possession-Only License No. DPR-34: Amendment revises the DTS.

Local Public Document Room location: Weld Library District - Downtown Branch, 919 7th Street, Greeley, CO 80631.

Sacramento Municipal Utility District, Docket No. 312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: February 28, 1995

Brief description of amendment: This amendment relocates the quality assurance audit frequencies from the technical specifications to the Rancho Seco Quality Manual and changes the reporting frequency of the Radioactive Effluent Release Report from semi-annual to annual.

Date of issuance: July 19, 1995

Effective date: July 19, 1995

Amendment No.: 122

Facility Operating License No. NPF-1: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16200)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 19, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Central Library, Government Documents, 828 I Street, Sacramento, California 95814.

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

Date of application for amendments: September 15, 1993, as supplemented by letter dated September 6, 1994.

Brief description of amendments: The amendments revised Technical Specification (TS) Table 2.2-1, ''Reactor Protective Instrumentation Trip Setpoint Limits,'' Table 3.3-1, ''Reactor Protective Instrumentation,'' Table 3.3-3, ''Engineered Safety Feature Actuation System Instrumentation,'' and Table 3.3-4, ''Engineered Safety Feature Actuation System Instrumentation Trip Values,'' and the associated Bases. The revisions to the notes in these tables change the pressure at which the low pressurizer pressure trip bypass shall be automatically removed to a consistent value of ''before pressurizer pressure exceeds 500 psia (the corresponding bistable allowable value is less than or equal to 472 psia).'' In addition, the wording of the notes is revised to make the notes more consistent with each other.

Date of issuance: July 14, 1995

Effective date: July 14, 1995, to be implemented within 30 days of the date of issuance.

Amendment Nos.: Unit 2 - Amendment No. 120; Unit 3 - Amendment No. 109

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

Date of initial notice in **Federal Register:** September 29, 1993 (58 FR 50975). The September 6, 1994, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 14, 1995. No significant hazards consideration comments received: No

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

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

Date of application for amendments: September 3, 1992

Brief description of amendments: These amendments revise TS 3/4.4.8 ''Pressure/Temperature Limits - Reactor Coolant System,'' and their associated Bases, following NRC guidance provided in Generic Letter 91-01, ''Removal of the Schedule for Withdrawal of Reactor Vessel Material Specimens from Technical Specifications.'' This generic letter allows licensees to remove the reactor vessel material surveillance capsule withdrawal schedules from the TS because they are a duplication of the requirements of 10 CFR Part 50 Appendix H.

Date of issuance: July 17, 1995

Effective date: July 17, 1995, to be implemented within 30 days of issuance

Amendment Nos.: Unit 2 - Amendment No. 121; Unit 3 - Amendment No. 110

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

Date of initial notice in **Federal Register:** February 17, 1993 (58 FR 8781)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: April 30, 1993, as supplemented by letters dated July 6, 1994 (separate letters for each unit), and letter dated January 27, 1995.

Brief description of amendments: These amendments revise TS 3/4.4.8.1, ''Pressure-Temperature Limits,'' TS 3.4.8.3.1, ''Overpressure Protection Systems-RCS Temperature less than or equal to °F [for Unit 2, less than or equal to 246°F for Unit 3],'' TS 3.4.8.3.2, ''Overpressure Protection Systems-RCS Temperature ≤256°F [for Unit 2, ≤246°F for Unit 3],'' and the associated TS Bases. The proposed change (1) revises the reactor coolant system (RCS) pressure-temperature (P-T) limits and the low temperature overpressure protection (LTOP) enable temperatures to be effective until 20 effective full power years (EFPY) of operation and (2) makes minor editorial changes.

Date of issuance: July 18, 1995

Effective date: July 18, 1995, to be implemented within 30 days of issuance

Amendment Nos.: Unit 2 - Amendment No. 122; Unit 3 - Amendment No. 111

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

Date of initial notice in **Federal Register:** Unit 2 - July 7, 1993 (58 FR 36445); Unit 3 - June 23, 1993 (58 FR 34094). The two supplemental letters dated July 6, 1994, and the January 27, 1995, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 30, 1994

Brief description of amendment: The amendments implement an analog transmitter/trip system on BFN Unit 3, revise the reactor vessel water level safety limit and limiting safety system setting for BFN Units 1 and 3, add instrument identifiers and revise calibration frequencies and functional test requirements for BFN Unit 2, revise the calibration frequency for instrumentation actuating the suppression chamber-reactor building vacuum breakers, and provide editorial changes.

Date of issuance: July 17, 1995

Effective date: July 17, 1995

Amendment Nos.: 222, 237, 196

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1994(59 FR 49435) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 17, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 356114.

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

Date of application for amendment: January 30, 1995

Brief description of amendment: The proposed amendment revises reactor coolant system pressure-temperature curves, changes bases for Technical Specification 3/4.4.9, Pressure Temperature Limits, and revises License Condition 2.C(3)(d) to reflect a change from 10 effective full power years (EFPY) to 21 EFPY.

Date of issuance: July 20, 1995

Effective date: July 20, 1995

Amendment No.: 199

Facility Operating License No. NPF-3. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14029)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 20, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendment: September 8, 1994

Brief description of amendment: The amendment revises Technical Specifications (TS) 4.2.2.2, 4.2.2.4, and 6.9.19. The changes address incorporating a penalty in the Core Operating Limits Report (COLR) to account for heat flux (F_Q) increases greater than 2 percent between measurements.

Date of issuance: July 20, 1995

Effective date: July 20, 1995

Amendment No.: 101

Facility Operating License No. NPF-30. Amendment revises the Technical Specification Surveillance Requirements and Administrative Controls.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65823). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 20, 1995. No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 22, 1994

Brief description of amendments: The amendments revised the Technical Specifications to delete unnecessary descriptive phrases regarding the number of cells in the station and emergency diesel generator batteries.

Date of issuance: July 11, 1995

Effective date: July 11, 1995

Amendment Nos.: 201 and 201

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18630)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 1995. No significant hazards consideration comments received: No

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

Dated at Rockville, Maryland, this 2nd day of August, 1995.

For the Nuclear Regulatory Commission

Jack W. Roe, 4Director, Division of Reactor Projects - III/IV, Office of Nuclear Reactor Regulation

[Doc. 95–18810 Filed 8–1–95; 8:45 am]

BILLING CODE 7590–01–F

[Docket No. 50–255]

Consumers Power Company; Notice of Partial Denial of Amendment to Facility Operating License and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) has partially denied a request by Consumers Power Company, (licensee) for an amendment to Facility Operating License No DRP–20 issued to the licensee for operation of Palisades, located in Covert Township, Van Buren County, Michigan. Notice of Consideration of Issuance of this amendment was published in the **Federal Register** on May 25, 1994 (59 FR 27053).

The purpose of the licensee's amendment request was to relocate certain Technical Specifications (TS) containing fuel cycle-specific parameter limits that can change with core reloads to a Core Operating Limits Report. Several of the TS bases have also been revised to refer to limits relocated to the COLR.

The NRC staff has concluded that the licensee's request cannot be fully granted. The removal of the power distribution measurement uncertainty factors in Table 3.23.3 and the addition of certain references to TS 6.9.1.f are denied. The licensee was notified of the Commission's denial of the proposed change by a letter dated July 26, 1995.

By September 1, 1995, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or 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.

A copy of any petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

For further details with respect to this action, see (1) the application for amendment dated April 7, 1994, as supplemented April 27, 1995, and (2) the Commission's letter to the licensee dated July 26, 1995.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, 20555 and at the Van Wylen Library, Hope College, Holland, Michigan 49423.

Dated at Rockville, Maryland, this 26th day of July 1995.

For the Nuclear Regulatory Commission.

Marsha Gamberoni,

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

[FR Doc. 95–18929 Filed 8–1–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 50–206, 50–361, 50–362]

In the Matter of Southern California Edison Company (San Onofre Nuclear Generating Station, Units 1, 2, and 3).

Southern California Edison Co.

Exemption

I

Southern California Edison Company (SCE or the licensee) is the holder of Facility Operating License No. DPR–13, which authorizes possession and maintenance of the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) and Facility Operating License Nos. NPF–10 and NPF–15, which authorizes operation of San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3), respectively. The licenses provide, among other things, that the SONGS units are subject to all rules, regulations, and orders of the Commission now or hereafter in effect. The facilities consist of three pressurized water reactors at the SCE site located in San Diego County, California. SONGS 1 is permanently shut down, while Unites 2 and 3 remain operational.

II

It is stated in 10 CFR 73.55, ''Requirements for physical protection of licensed activities in nuclear power reactors against radological sabotage,'' paragraph (a), that ''The licenses shall establish and maintain an onsite physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.''

It is specified in 10 CFR 73.55(d), ''Access Requirements,'' paragraph (1) that ''The licensee shall control all points of personnel and vehicle access into a protected area.'' It is specified in 10 CFR 73.55(d)(5) that ''A numbered