Safeguards Policy and Procedures Letter 1–50, Revision 1, is not warranted.

Alternatives to the Proposed Action

The proposed action is to amend NRC Source Material License SUA–648, for reclamation of the Heap Leach Area, as requested by Umetco. Therefore, the principal alternatives available to NRC are to:

1. Approve the license amendment request as submitted; or
2. Amend the license with such additional conditions as are considered necessary or appropriate to protect public health and safety and the environment; or
3. Deny the amendment request.

Based on its review, the NRC staff has concluded that the environmental impacts associated with the proposed action do not warrant either the limiting of Umetco's future operations or the denial of the license amendment. Additionally, in the TER prepared for this action, the staff has reviewed the licensee's proposed action with respect to the criteria for reclamation, specified in 10 CFR Part 40, Appendix A, and has no basis for denial of the proposed action. Therefore, the staff considers that Alternative 1 is the appropriate alternative for selection.

Finding of No Significant Impact

The NRC staff has prepared an EA for the proposed renewal of NRC Source Material License SUA–648. On the basis of this assessment, the NRC staff has concluded that the environmental impacts that may result from the proposed action would not be significant, and therefore, preparation of an Environmental Impact Statement is not warranted.

The EA and other documents related to this proposed action are available for public inspection and copying at the NRC Public Document Room, in the Gelman Building, 2120 L Street N.W., Washington, DC 20555.

Notice of Opportunity for Hearing

The Commission hereby provides notice that this is a proceeding on an application for a licensing action falling within the scope of Subpart L, ''Informal Hearing Procedures for Adjudications in Materials and Operators Licensing Proceedings,'' of the Commission's Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders in 10 CFR Part 2 (54 FR 8269). Pursuant to § 2.1205(a), any person whose interest may be affected by this proceeding may file a request for a hearing. In accordance with § 2.1205(c), a request for a hearing must be filed within thirty (30) days from the date of publication of this **Federal Register** notice. The request for a hearing must be filed with the Office of the Secretary either:

(1) By delivery to the Rulemakings and Adjudications Staff of the Office of the Secretary at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; or

(2) By mail or telegram addressed to the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Rulemakings and Adjudications Staff.

Each request for a hearing must also be served, by delivering it personally or by mail to:

(1) The applicant, Umetco Mineral Corporation, P.O. 1029, Grand Junction, CO 81502;

(2) The NRC staff, by delivery to the Executive Director of Operations, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, or

(3) By mail addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

In addition to meeting other applicable requirements of 10 CFR Part 2 of the Commission's regulations, a request for a hearing filed by a person other than an applicant must describe in detail:

(1) The interest of the requestor in the proceeding;

(2) How that interest may be affected by the results of the proceeding, including the reasons why the requestor should be permitted a hearing, with particular reference to the factors set out in § 2.1205(g);

(3) The requestor's areas of concern about the licensing activity that is the subject matter of the proceeding; and

(4) The circumstances establishing that the request for a hearing is timely in accordance with § 2.1205(c).

Any hearing that is requested and granted will be held in accordance with the Commission's ''Informal Hearing Procedures for Adjudications in Materials and Operator Licensing Proceedings'' in 10 CFR Part 2, Subpart L.

Dated at Rockville, Maryland, this 30th day of April 1998.

For the Nuclear Regulatory Commission.

Joseph J. Holonich,

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

[FR Doc. 98–11980 Filed 5–5–98; 8:45 am]

BILLING CODE 7590–01–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. Pub. L. 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 April 10 through April 24, 1998. The last biweekly notice was published on April 22, 1998 (63 FR 19964).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period.

However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By June 5, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, 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 the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: November 1, 1996, as supplemented by letters dated October 13, 1997, February 26, 1998, and March 13, 1998.

Description of amendment request: Associated with a Carolina Power & Light Company (the licensee) application to convert from the Current Technical Specifications (CTS) for the Brunswick Steam Electric Plant, Units 1 and 2, to Improved Technical Specifications (ITS), as contained in Revision 1 of NUREG–1433, ''Standard Technical Specification General Electric Plants, BWR/4,'' the licensee proposed removing a restriction on a surveillance test described below.

CTS 4.8.1.1.1.b requires that the offsite electrical power circuits be demonstrated OPERABLE, at least once per 18 months during shut down, by manually transferring the unit power supply from the normal circuit to the alternate circuit. As proposed, ITS SR 3.8.1.8.b will not contain the restriction to perform the Surveillance ''during shutdown.'' Currently, this test is performed by momentarily paralleling the 230 kV offsite alternating current (AC) power sources. The licensee has stated that paralleling offsite AC power sources is a controlled evolution and the increased risk associated with the performance of this test while the unit is at power is not significant for the following reasons: (1) the frequency and voltages are verified to be within the required range prior to paralleling the two offsite AC power sources; (2) breaker interlocks ensure that the alternate circuit is connected to the load prior to opening the preferred circuit; (3) the test does not result in de-energization of any 4.16 kV emergency bus and the potential for electrical perturbations on the grid system is the same whether performing the transfer while the unit is at power or while shutdown; and (4) operating history indicates that transferring offsite AC power sources while the units were in Operational Conditions 1 (power operation) or 2 (startup) has been performed satisfactorily without electrical distribution system perturbations. The licensee has further pointed out that Generic Letter 91–04, ''Changes in Technical Specifications to Accommodate a 24-Month Fuel Cycle,'' states that licensees may omit the Technical Specification qualification that a refueling interval surveillance is to be performed ''during shutdown.'' Therefore, consistent with the guidance provided in Generic Letter 91–04, the licensee proposed deletion of the requirement to perform this Surveillance ''during shutdown'' as part of the conversion from CTS 4.8.1.1.1.b to ITS SR 3.8.1.8.b.

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?

This change would remove a specific restriction to perform the verification of the manual transfer of the unit power supply from the normal circuit to the alternate circuit ''during shutdown.'' The transfer of the unit power supply from the normal circuit to the alternate circuit is not an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. Currently, this test is performed by momentarily paralleling the 230 kV offsite AC power sources. Paralleling offsite AC power sources is a controlled evolution and the increased risk associated with the performance of this test while the unit is at power is not significant for the following reasons: (1) The frequency and voltages are verified to be within the required range prior to paralleling the two offsite AC power sources; (2) breaker interlocks ensure that the alternate circuit is connected to the load prior to opening the preferred circuit; (3) the test does not result in de-energization of any 4.16 kV emergency bus and the potential for electrical perturbations on the grid system is the same whether performing the transfer while the unit is at power or while shutdown; and (4) operating history indicates that transferring offsite AC power sources while the units were in MODE (Operational Condition) 1 or 2 has been performed satisfactorily without electrical distribution system perturbations. The appropriate plant conditions for performance of the Surveillance will continue to be controlled to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in Generic Letter 91–04. Therefore, this change does not significantly increase the consequences of any previously analyzed accident.

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

This change removes a specific restriction on the plant conditions for performing a Surveillance, but does not change the method of performance. The appropriate plant conditions for performance of the Surveillance will continue to be controlled to assure the possibility for a new or different kind of accident are not created. This control method has been previously determined to be acceptable as indicated in Generic Letter 91–04. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

The margin of safety considered in determining the appropriate plant conditions for performing the Surveillance will continue to be controlled to assure that there is no significant reduction. This control method has been previously determined to be acceptable as indicated in Generic Letter 91–04. Therefore, the 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: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Pao-Tsin Kuo

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

Date of amendment request: April 3, 1998.

Description of amendment request: The Carolina Power & Light Company, licensee for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, proposed amendments to the Technical Specifications (TS) to change the specified total volume of the condensate storage tank (CST) from 150,000 gallons to 228,200 gallons. During a recent review of industry operating experience, the licensee determined that information contained in TS 3.5.3.1, Core Spray System (CSS), and the associated bases regarding water inventory in the CST was incorrect. Specifically, the minimum CST volume requirement contained in TS 3.5.3.1 would not assure the availability of 50,000 gallons of water for the CSS, as indicated in TS Bases section 3/4.5.3.1 for the CSS.

The licensee has concluded that the proposed license amendments do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10 CFR 50.92 is provided below.

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

The proposed TS change revises the minimum CST [Condensate Storage Tank] water volume required for OPERABILITY of the Core Spray system (CSS) in OPERATIONAL CONDITIONS 4 AND 5 when the suppression pool is inoperable. The proposed change does not alter the operation of any plant system or component; does not involve a physical modification to any structure, system, or component; and does not affect an initiator to any accident previously evaluated. The minimum CST water level is being increased to assure the availability of 50,000 gallons of water for use by the CSS. Therefore, the proposed license amendments do not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed TS change revises the minimum CST water volume required for OPERABILITY of the CSS in OPERATIONAL CONDITIONS 4 and 5 when the suppression pool is inoperable. The proposed change does not alter the operation of any plant system or component; does not involve a physical modification to any structure, system, or component; and does not affect an initiator to any accident previously evaluated. The proposed change does not add or modify equipment or components related to the CSS and will, therefore, not create new failure modes or common failure modes. The minimum CST water level is being increased to assure the availability of 50,000 gallons of water for use by the CSS. Therefore, the proposed license amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. The proposed license amendments increase the minimum CST water level to assure the availability of 50,000 gallons of water for use by the CSS. These volumes ensure the validity of existing analyses, and ensure that the existing TS Bases are satisfied. The proposed change does not involve a physical modification to any structure, system, or component, and does not modify the operation of any existing equipment. Therefore, the proposed license amendments do not involve a reduction in a margin of safety.

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

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

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Pao-Tsin Kuo (Acting)

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: March 31, 1998.

Description of amendment request: Unreviewed Safety Question involving use of Station Blackout (SBO) diesel generators (DGs) and use of a mobile safe shutdown (SSD) battery cart in the 10 CFR part 50, appendix R, Safe Shutdown Safety Analysis.

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 licensee has provided a separate no significant hazards consideration determination for the SBO DGs and the battery cart under this amendment request. The following is the determination for the SBO DGs:

(1) No significant increase in the probability or consequences of an accident previously evaluated is involved because of the following:

Two types of previously evaluated accidents are relevant to this criterion: (1) A fire; (2) other accident evaluated in the UFSAR. For these previously evaluated accidents, the change would not result in an increase in either their probabilities of occurrence or the consequences of their occurrence, for the following reasons.

The use of the SBO DGs in lieu of the [Emergency Diesel Generators] EDGs does not change the probability or consequences of a fire. The likelihood of a fire is unchanged. Use of the SBO DGs does not significantly change the fire loading nor introduce significant new ignition sources. The consequences of a fire are unchanged because use of the SBO DGs continues to support the station's ability to achieve and maintain shutdown in the event of a fire.

Use of the SBO DGs for non-fire purposes is unchanged by use of the SBO DGs for post-fire safe shutdown in the event of a fire in areas requiring alternate shutdown capability. Accordingly there is no change in the probability or consequences of a previously evaluated accident involving the SBO DGs. Similarly, there is no change to the probability or consequences of other accidents that have been previously evaluated because they are independent of this change in use of the SBO DGs.

(2) The possibility of a new or different kind of accident from any accident previously evaluated is not created because:

The proposed change does not create the possibility of a new or different kind of accident from that previously evaluated for Quad Station. Although the SBO DGs will be used for a new function, there is no significant change in the operation of the SBOs for a non-fire event. Moreover, the overall use of the SBO DGs as an AC power source is not significantly different from the use of the EDGs. The SBO DGs buses provide power to the same buses that are powered from the EDGs. No new modes of operation are introduced by the proposed changes. The use of the SBO DGs provides a slightly different but effective method for achieving and maintaining post-fire safe shutdown for areas requiring alternate shutdown capability. As such, the proposed change does not create the possibility of a new or different kind of accident.

(3) No significant reduction in the margin of safety is involved because:

A change in the fire protection program does not result in a significant reduction in the margin of safety if the change does not result in a significant adverse impact on the plant's ability to achieve and maintain safe shutdown in the event of a fire. The proposed use of the SBO DGs instead of the EDGs to achieve and maintain safe shutdown within 72 hours change does not significantly affect the capability or reliability of the equipment assumed to operate in the safety analysis.

The demonstrated capability and reliability of the SBO and EDGs are not significantly different. Indeed, the SBO DGs represent a safety improvement due to their physical separation from the postulated fire areas, and the operational benefits provided by their greater capacity. Any narrow reduction in margin associated with the need to manually start the SBO DGs is offset by the reduction in manual actions necessary to reduce electrical loads powered from the EDGs. The lack of Class 1E qualification for the SBO DGs is not significant from a safety perspective because the demonstrated reliability of the SBO DGs is comparable to the reliability of the EDGs. The lack of seismic qualification and single failure protection do not constitute a significant reduction in margin since neither of these attributes is required by Appendix R. Accordingly, the Commission has already determined that these attributes are not part of the Appendix R acceptance criterion. Any reduction in margin associated with the greater fuel consumption rate of the SBO DGs is partially offset by the increased flexibility in powering equipment to achieve and maintain post fire safe shutdown. Additionally, onsite fuel storage and manual transfer capabilities provide for at least 72 hours of SBO DG operation. Within 72 hours, deliveries of diesel fuel from offsite supplies is expected. Therefore, the use of the SBO DGs as an onsite AC power source for equipment necessary to achieve and maintain post-fire safe shutdown in areas requiring alternate capabilities does not involve a significant reduction in margin.

The licensee has evaluated the use of the mobile SSD battery cart to provide the power source for the Automatic Depressurization System (ADS) valves under certain scenarios where the valves are needed to achieve cold shutdown and determined that it does not involve a significant hazards consideration for the reasons discussed below.

(1) No significant increase in the probability or consequences of an accident previously evaluated is involved.

The accident previously evaluated is the postulated fire requiring alternate shutdown capability. The probability of a previously evaluated fire is not increased significantly because the mobile SSD batteries do not create significant new ignition sources or any other fire initiators. The consequences of a previously evaluated fire are not increased significantly because the mobile SSD batteries do not significantly increase the fire loading in the plant, do not interfere with the plant's ability to extinguish a fire, and are fully capable of fulfilling the designed safety function.

The associated systems related to this proposed change are not affected in a way that could impact the initiation of any accident sequence for the Quad Cities Station. No modes of operation are introduced by the proposed change such that adverse consequences result.

The probability of an accident involving the use of the mobile SSD batteries would not be increased significantly by this proposed use because the use is not significantly different from the alternative manual attachment of a power source to the ADS valves.

The consequences of an accident involving the use of the mobile SSD batteries are not increased because the only significant consequences would be a delay in achieving cold shutdown and that would have no different consequences than would a delay due to an accident related to the currently used manual power source.

(2) The possibility of a new or different kind of accident from any accident previously evaluated is not created.

The proposed change for the Quad Cities Station does not create the possibility of a new or different kind of accident from that previously evaluated. Because the mobile SSD batteries simply provide a different form of manually connecting a source of power to the ADS valves, the use of the mobile SSD batteries does not present new or different kinds of accidents related to such manual actions. Finally, because no new modes of operation are introduced by the proposed change, the change does not create the possibility of a new or different kind of accident that could be related to new modes of operation.

(3) No significant reduction in the margin of safety is involved.

The analytic framework for determining the extent to which a proposed change affects the margin of safety has been discussed above and, so will not be repeated here. In this case, a review of the proposed changes shows that they will not have an adverse impact on the ability to achieve and maintain safe shutdown. Several features associated with the use of the mobile SSD batteries show, as discussed above, that it provides an effective method for achieving and maintaining safe shutdown following a fire. In particular, use of the mobile SSD batteries reduces the overall complexity of the cold shutdown repairs required to supply power to the ADS valves and is familiar to plant personnel from their training on its use for other purposes.

Design calculations regarding capabilities of the mobile SSD batteries show they will be capable in fulfilling their intended safety function for their design basis Appendix R scenario. Reliability of the mobile SSD batteries will be maintained by augmented quality standards. This will entail the conduct of appropriate maintenance and surveillance which is designed to ensure that the mobile batteries will function as intended. Reliability of this power source is further enhanced by the circumstance that there are two mobile SSD batteries, thus permitting one to act as a backup to the other.

Under these circumstances, the margin of safety for achieving cold shutdown using the ADS valves is not reduced significantly, if at all, by the use of non-safety related mobile SSD batteries to power the ADS valves. Although safety-related station batteries had previously been used in this function, the method for attaching those batteries was more prone to human error than the method which has been developed for the mobile SSD batteries. Moreover, substantial steps have been taken to provide a high level of reliability for the mobile SSD batteries. Overall, therefore, the ability to achieve and maintain safe shutdown in the event of a fire has not been reduced by this change in the source of power to the ADS valves.

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 are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

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

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

NRC Project Director: Stuart A. Richards

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

Date of amendment request: March 30, 1998.

Description of amendment request: The proposed amendments would restore the Zion Custom Technical Specifications (CTS) that had been replaced with Improved Technical Specification by a previous amendment and would reinstate License Conditions that were deleted by that previous amendment. The proposed amendment would also modify the CTS to allow the use of Certified Fuel Handlers to satisfy shift staffing requirements and would change management titles and responsibilities to reflect the permanently shutdown organization.

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:

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

With a plant permanently shutdown and defueled the spectrum of accidents and events that remain credible is significantly reduced. As discussed below the proposed changes do not affect the probability or consequences of any accidents that do remain credible.

The restoration of the CTS which were replaced with the ITS by Amendments 178/165 cannot increase the probability or consequences of any event or accident because the amendment was never implemented. The CTS have been maintained as the legally binding Technical Specifications in effect at Zion Station. The reinstatement of the five License Conditions deleted by Amendments 178/165 is an administrative change in that the requirements contained in the License Conditions had been relocated elsewhere and are now being restored exactly as they were before the amendment was issued. Since the actual requirements have not changed there can be no change in the probability or consequences of any accident or event.

The changes in management titles and responsibilities will not increase the probability or consequences of any accident or event because these changes are administrative and will not result in any decrease in the quality of management applied to Zion Station. The changes are commensurate with the significant reduction in site activities, site staffing, and risk to public health and safety that occurs when an operational nuclear power plant transitions to a permanently shutdown and defueled plant. Responsible individuals will have the authority to commit the personnel and resources necessary to fulfill their obligations for safe storage and handling of nuclear fuel. The change of position designations will have no effect on the frequency of occurrence of accident or event initiators, or on their consequences.

The changes to allow use of Certified Fuel Handlers in lieu of personnel licensed in accordance with 10 CFR part 55 will not increase the probability or consequences of an accident or event because the Certified Fuel Handler Training and Retraining program (which will be approved by the

NRC) has been developed using a Systems Approach to Training as defined in 10 CFR 55.4. This approach provides assurance that the Certified Fuel Handlers have the knowledge, skills, and abilities that are commensurate with the tasks to be performed (i.e., the proper monitoring, handling, storage, and cooling of nuclear fuel). Therefore the frequency of occurrence of accident or event initiators is not increased and the consequences of the accidents or events are unaffected.

The changes in shift staffing numbers and crew composition will not increase the probability or consequences of an accident or event. These staffing changes are commensurate with the quantity, complexity, and hazard level of the activities required for storage and handling of nuclear fuel. The elimination of the Shift Control Room Engineer does not affect any accident or event initiator or consequence since the previous specification would not have required that the position be manned with both units shut down. The elimination of the requirement for a Radiation Protection Person on shift will have no effect on the frequency of occurrence of accidents or events, nor on the consequences of the accident or event.

The changes in verbiage to eliminate any implication that units are operational will not increase the probability or consequences of an accident or event because they are largely editorial changes and do not increase the frequency of occurrence of [or] event initiators, nor do they increase the consequences.

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

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

The changes proposed by this amendment do not involve new structures, systems, or components, or the use of existing structures, systems, or components in a new manner. Consequently no new failure mechanisms are introduced. The design and operation of structures, systems, or components is unaffected by:

The restoration of CTS,

The reinstatement of the five License Conditions deleted by Amendments 178/165,

The changes in management titles and responsibilities,

The changes to allow use of Certified Fuel Handlers in lieu of 10 CFR [Part] 55 licensed personnel,

The changes in shift staffing numbers and crew composition, or

The changes in verbiage to eliminate any implication that units are operational.

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

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

One of the License Conditions that would be reinstated by this amendment establishes limits that help ensure that the assumptions of the fuel handling accident analysis remain valid. License Condition 2.C.(7).b limits the weight of loads carried over fuel stored in the spent fuel pool to the weight of a single fuel assembly plus the tool for moving that assembly. This weight limit ensures that the number of fuel rods broken in a fuel handling accident does not exceed the maximum number of fuel rods assumed to break in the accident analysis. Consequently, this change continues to provide assurance that the margin of safety involving the number of fuel rods broken in the accident will not be reduced.

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

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

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

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

NRC Project Director: Stuart A. Richards

Duke Energy Corporation (DEC), et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: May 27, 1997, as supplemented by a letter dated April 20, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) of each unit to conform with NUREG–1431, Revision 1, ''Standard Technical Specifications—Westinghouse Plants.'' The Commission had previously issued a Notice of Consideration of Issuance of Amendments in the **Federal Register** on July 14, 1997 (62 FR 37628) covering all the proposed changes that were indeed within the scope of NUREG–1431. In DEC's May 27, 1997, submittal, there are proposed changes that are beyond the scope of NUREG–1431, which were thus not covered by the staff's July 14, 1997, notice. The following descriptions and no significant hazard analyses cover only those beyond-scope changes. Associated with each change are administrative/editorial changes such that the new or revised requirements would fit into the format of NUREG–1431.

1. This proposed change affects the surveillance requirement currently contained in Sections 4.6.6.1 and 4.6.6.2, regarding the containment valve injection water system. The requirement to assure adequate capacity to maintain system pressure for at least 30 days would be deleted, the required system pressure of 16.2 pounds per square inch gauge (psig) would be replaced with a surge tank pressure of 36.4 psig, and the system would be tested at lower pressures and more restrictive leak rates.

2. Section 3.9.2.1, regarding the boron dilution mitigating system, currently requires both trains to be operable in Mode 6 (refueling). DEC proposed to add a note stating that the system may be blocked during core reloading until two assemblies are loaded into the core. Adequate shutdown margin will continue to be controlled and verified by other specifications. This blocking would prevent inadvertent actuation of the system, which could distract the operating personnel, but would not diminish the monitoring function of the system.

3. DEC proposed to change the definition of 'dose equivalent iodine-131.' Subsequently, this proposed change was withdrawn by letter dated April 20, 1998.

4. DEC proposed to change Section 3.3.3.6 regarding accident monitoring instrumentation. Specifically, the change would (a) increase the time allowed to return the required number of channels to operable; and (b) permit continued operation if one channel is inoperable given certain conditions are met, instead of requiring shutdown.

5. DEC proposed to change Section 4.6.4.1 regarding surveillance requirements for the hydrogen monitors (combustible gas control). Specifically, this would eliminate the channel operational test, and extend the channel check frequency from once per 12 hours to once per 31 days.

6. DEC proposed to change Section 3.4.6.1 regarding reactor coolant leakage detection systems; a system comprising diverse instruments such as gaseous radioactivity monitoring, containment floor and equipment sump monitoring, etc. In addition to the instruments specified by this section, the plant has other installed instruments such as monitors for humidity, temperature, etc., which can provide indication for reactor coolant leakage. Currently, this specification allows operation up to 30 days if the containment floor and equipment sump monitoring system is inoperable. The change would impose a requirement to perform a precision water balance of the reactor coolant system every 24 hours during this period. The change would also reduce the number of monitors required operable provided compensatory measures are performed or diverse instruments continue to be available.

7. DEC proposed to change Section 4.5.4.b, which currently requires verification of the refueling water storage tank temperature to be within the allowed range once per 24 hours if the outside air temperature is less than 70 degrees or greater than 100 degrees Fahrenheit. The proposed change would simply require that the tank temperature be verified within range every 24 hours regardless of outside air temperature.

8. DEC proposed to revise Table 3.7–1, which imposes limits on the maximum allowable power range neutron flux high setpoint for various numbers of inoperable safety valves on any operating steam generator. The revision would reduce the setpoints, making them more conservative.

9. Section 3.7.6, regarding the condensate storage system, currently only exists in the Unit 2 TS. DEC proposed to impose these requirements also on Unit 1.

10. Several electrical busses and inverters currently covered by Section 3.8.3.1 are qualified by a footnote, which specifies the conditions under which the inverter may be disconnected from its direct current source. DEC proposed to delete this footnote because it is not needed.

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

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

For changes 1, 2, 4, 5, 6, 7, 8, 9, and 10, the answer is ''no.'' The proposed changes will not affect the safety function of the subject systems. There will be no direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of these systems, and the systems were not factors in the consequences of previously analyzed accidents. Therefore, the proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

For changes 1, 2, 4, 5, 6, 7, 8, 9, and 10, the answer is ''no.'' The proposed changes would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

For changes 1, 2, 4, 5, 6, 7, 8, 9, and 10, the answer is ''no.'' Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

NRC Project Director: Herbert N. Berkow

Duke Energy Corporation (DEC), et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: April 8, 1998.

Description of amendment request: The proposed amendments would revise Section 3.6.5.1 and 4.6.5.1 of the Technical Specifications (TS) of each unit to relax ice condenser stored ice weight requirements by approximately 6 percent. The proposed change is based mainly on DEC's gathered data showing lower sublimation rate than originally anticipated.

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

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

No. The proposed changes will not affect the safety function of the ice condenser in that there will be no changes to the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of the ice condenser, and the ice condenser will remain fully capable of performing its design accident mitigation function. Therefore, the proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

2. Will the changes create the possibility of a new or difference kind of accident from any accident previously evaluated?

No. The proposed changes would not lead to any hardware or operating procedure change. Reducing the required ice weight will not have any impact on other plant systems that were assumed to be accident initiators. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

3. Will the changes involve a significant reduction in a margin of safety? No. Margin of safety is associated with confidence in the design and operation of the plant; specifically, the ability of the fission product barriers to perform their design functions during and following an accident. The proposed changes regarding required ice weight do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

NRC Project Director: Herbert N. Berkow

Duke Energy Corporation (DEC), Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: May 27, 1997.

Description of amendment request: The proposed changes would lower the minimum required diesel generator (DG) air start receiver pressure from 220 per square inch gauge (psig) to 210 psig with a monthly verification, and would include an allowed outage time of 48 hours for a degraded air receiver provided the redundant air receiver is maintained at equal to or greater than 210 psig. These proposed changes are associated with DEC's application to convert to the Improved Technical

Specifications. Also, they are considered less restrictive requirements because of the lower required minimum pressure and the allowance of continued operation with a degraded starting air system.

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

1. (Do the changes) involve a significant increase in the probability or consequence of an accident previously evaluated?

The proposed changes provide Actions for degraded capabilities of the diesel starting air subsystems for the DG. The proposed Actions establish limits for the DG starting air subsystems of 210 psig, (are) allowed to decrease below the required value for 48 hours(, and are verified every 31 days.) The Completion Times are based on the amount of capability remaining, and the time needed to correct any deficient condition. If the Completion Times are exceeded, the specification requires the associated DG to be declared inoperable immediately, consistent with the current TS (technical specifications). Since the new Actions continue to assure that the associated DG remains capable of performing its design safety function, the proposed (changes do) not significantly affect the probability or consequences of an accident previously evaluated.

2. (Do the changes) create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed (changes do) not permit operation in a new or different mode, or permit the installation of a new or different type of equipment. The proposed changes provide Actions for degraded capabilities of the DG starting air subsystems. The proposed Actions establish Conditions, Required Actions, and Completion Times to be entered when in a degraded condition. The DG remains capable of performing its design safety function. Therefore, the proposed (changes do) not create the possibility of a new or different kind of accident from those previously evaluated.

3. (Do these changes) involve a significant reduction in a margin of safety?

The proposed (changes do) not significantly increase the probability or consequences of an accident previously evaluated. The changes provide assurance that timely action will be initiated to restore DG starting air subsystem when inoperabilities exist, without unnecessarily forcing plant shutdown. Based on the limit for the starting air subsystem for the DG, the limited time allowed is acceptable to restore the parameter to within the requirements without unnecessary plant shutdown. Therefore, (these 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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

NRC Project Director: Herbert N. Berkow

Duke Energy Corporation (DEC), Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: May 27, 1997.

Description of amendment request: The two proposed changes are associated with DEC's application to convert to the Improved Technical Specifications and are considered as administrative changes. The first change would delete a current requirement to only verify the refueling water storage tank temperature once every 24 hours if the outside air temperature is less than 70 degrees or greater than 100 degrees Fahrenheit, and would require that the tank temperature be verified within range every 24 hours regardless of the outside air temperature value. The second change would delete the current requirement that 32 of 33 hydrogen igniters be operable on each train, and would require that 34 igniters per train to be operable. The actual design contains 35 igniters per train. This change would correct an inadvertent error in the current Technical Specifications (TS). The number of igniters was increased to 35 after the first refueling outage of each unit. This change would correct the TS to reflect the requirements stated in Safety Evaluation Report Supplement 7.

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 for each of the above proposed changes. The NRC staff has reviewed the licensee's analyses against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

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

The proposed changes will not affect the safety function of the subject systems. There will be no direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of these systems, and the systems were not factors in the consequences of previously analyzed accidents. Therefore, the proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

The proposed changes would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: May 27, 1997.

Description of amendment request: The proposed change would allow two charging pumps or safety injection pumps capable of injecting into the Reactor Coolant System (RCS) when the RCS is depressurized and an RCS vent of at least 4.5 square inches is established. This proposed change is associated with the licensee's application to convert to the Improved Technical Specifications and results in a requirement less restrictive than the current requirement.

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

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

The proposed change will provide an additional alternative for low temperature (overpressure) relief capacity when two charging pumps or safety injection pumps are capable of injecting into the RCS. The low temperature (overpressure) protection is not considered to be an initiator of any analyzed event, therefore, the proposed change does not increase the probability of a previously analyzed event.

The proposed change provides an equivalent vent size to the existing two open PORVs (power-operated relief valves). 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the manner in which the plant is operated. The proposed change adds an additional alternative to overpressure protection equivalent to the current requirements. Therefore, the proposed change will not create the possibility of a new or different kind of accident than any previously evaluated.

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

As described above, the proposed change adds an additional alternative to overpressure protection equivalent to the current requirements. The inclusion of additional alternatives provides the operating staff with additional flexibility in meeting low temperature overpressure protection requirements. Therefore, the 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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina

NRC Project Director: Herbert N. Berkow

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: March 25, 1998

Description of amendment request: Revise Technical Specification (TS) 3.9.8.1, ''Shutdown Coolant and Coolant Circulation High Water Level,'' and TS 3.9.8.2, ''Shutdown Cooling and Coolant Circulation Low Water Level,'' to change the minimum water level above the fuel assemblies seated in the reactor vessel at which the Shutdown Cooling (SDC) System is required to be maintained operable, or be in operation. In addition, TS 3.8.1.2, ''Electric Power Systems, A.C. Sources, Shutdown,'' and Technical Specification Bases 3/4.9.8, ''Shutdown Cooling and Coolant Circulation,'' have been changed to make the wording consistent with TS 3.9.8.1 and TS 3.9.8.2.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequence of any accident?

Response: No.

The operation of the facility in accordance with this change does not involve an increase in the probability of any accident.

Changing the water level at which the Shutdown Cooling (SDC) System is required to be maintained operable or be in operation will not increase the probability or consequences of an accident. The design, operation, or configuration of the SDC system will not be changed.

At least one shutdown cooling train will be in operation to ensure sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140 degree F as required during the refueling mode.

At least one shutdown cooling train will be in operation to ensure sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. Technical Specification 3.9.10.1, ''Refueling Operations Water Level—Reactor Vessel Fuel Assemblies,'' will be complied with, and therefore, the assumptions related to iodine removal and the fuel handling accident will be preserved.

Sufficient time, approximately 1.00 hours, will be available to the operators to initiate compensatory measures to preclude the initiation of core boiling in the unlikely event SDC should be loss [lost].

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not affect the design, configuration, or operation of the SDC system, and therefore there are no new modes of failure introduced.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

Operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

The calculation of the time to the initiation of boiling based on 23 feet above the top of the fuel seated in the reactor vessel, at four days after shutdown, demonstrates there is significant time available, approximately 1.00 hour, to the operators within which to take compensatory measures to preclude the initiation of boiling. The calculation shows that based on 23 feet of water above the reactor flange there is 2.04 hours to the initiation of boiling. Although there is a reduction in the time to the initiation of boiling, compensatory measures could be taken within a few minutes to restore SDC, and thus, there is still a significant margin available to the operators within which to preclude the initiation of boiling. Thus, the margin of safety is not significantly reduced.

The time to core uncovery was determined to be 27.74 hours based on four days after shutdown and water level twenty-three (23) feet above the fuel assemblies seated in the reactor vessel.

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 are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit No. 3, Citrus County, Florida

Date of amendment request: March 20, 1998.

Description of amendment request: The proposed amendment requests editorial changes to the Improved Technical Specifications (ITS) Safety Limits and Administrative Controls to replace the titles of the Senior Vice President, Nuclear Operations (SVPNO) and the Vice President, Nuclear Production (VPNP) with the position of Chief Nuclear Officer (CNO). The CNO combines the duties of the SVPNO and VPNP as currently described in ITS and is required to be an officer of the company. The proposed change is intended to allow upgrading the position of the corporate officer responsible for overall nuclear operations without limiting the title.

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.

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because the deletion and updating of individual titles does not affect plant operation. No design basis accidents are affected by the proposed administrative and editorial changes and, as such, there are no physical changes to the facility or its operation.

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

The proposed ITS changes are administrative and editorial in nature. No changes to the facility structures, systems and components or their operation will result. The design and design basis of the facility remain unchanged. The plant safety analyses remain current and accurate. No new or different failure mechanisms are introduced. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not introduced.

Does not involve a significant reduction in the margin of safety.

The proposed ITS changes are administrative and editorial in nature. The proposed safety margins established through the design and facility license including the Improved Technical Specifications remain unchanged. In addition, the proposed amendment ensures continued emphasis and assignment of responsibility for overall nuclear safety. Therefore, all margins of safety are maintained.

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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042

NRC Project Director: Frederick J. Hebdon

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit No. 3, Citrus County, Florida

Date of amendment request: March 20, 1998.

Description of amendment request: The proposed amendment would change the Inservice Inspection Program described in Improved Technical Specification (ITS) 5.6.2.8.c. This ITS currently states that the reactor coolant pump (RCP) motor flywheels will be inspected during the ''Spring 1998 refueling outage,'' which would have been refueling outage 11. Due to a recent 17-month extended outage, refueling outage 11 has been deferred until Fall 1999. The proposed change is intended to accurately reflect the new refueling outage 11 schedule.

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

The proposed change will not significantly increase the probability or consequences of an accident previously evaluated.

The safety function of the RCP flywheels is to provide a coastdown period during which the RCPs would continue to provide reactor coolant flow to the reactor after loss of power to the RCPs. The maximum loading on the RCP motor flywheel results from overspeed following a large loss of coolant accident (LOCA). The estimated maximum obtainable speed in the event of a Reactor Coolant System piping break was established conservatively. The proposed one-time editorial change to remove the words ''Spring 1998 refueling outage'' and replace them with ''to coincide with Refueling Outage 11R'' does not affect that analysis. The proposed change in dates is editorial in that it merely reflects the new date for cycle 11. The usage time for the flywheels is bounded by the original estimates. The proposed editorial change does not affect the amount of radioactive material available for release or modify any systems used for mitigation of such releases during accident conditions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

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

The proposed change will not involve a significant reduction to any margin of safety.

The proposed Amendment is an editorial change to reflect that CR–3's operating cycle is not ending in spring 1998, but in fall 1999. The proposed change does not affect the methods of inspection or its acceptance criteria. Therefore, the margins of safety defined in RG [Regulatory Guide] 1.14 are not changed.

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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: April 15, 1998.

Description of amendment request: The proposed amendment would update the existing pressure-temperature curves with new curves with values from 18 to 32 effective full power years based on the testing and analysis of reactor pressure vessel surveillance materials.

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 amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The pressure-temperature limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the ASME B&PV Code and 10 CFR part 50 appendices G and H as restrictions on normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.

(2) The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The amendment will merely update the pressure-temperature curves (and associated SRs and Bases) already existing in the plant Improved Technical Specifications to provide limits from 18 to 32 EFPY of operation, which are based upon evaluation and analysis of actual in-vessel material specimens, per 10 CFR part

50, appendices G and H. The pressure-temperature curves are established to the requirements of 10 CFR part 50, appendix G to assure that brittle fracture of the reactor vessel is prevented.

(3) The proposed amendment will not involve a significant reduction in a margin of safety. 10 CFR part 50, appendix G specifies fracture toughness requirements to provide adequate margins of safety during operation over the service lifetime. The values of adjusted reference temperature and upper shelf energy determined as a result of the 10 CFR part 50, appendices G and H analysis are expected to remain within the limits of Regulatory Guide 1.99, Revision 2 and appendix G of 10 CFR part 50 (less than 200° F and greater than 50 ft-lbs respectively) for at least 32 EFPY of operation.

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

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, IA 52401

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

Acting NRC Project Director: Richard P. Savio

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

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from the current Technical Specifications (TS) to improved TS, would establish Allowable Values for the instrumentation included in Section 3.3, as a result of the plant-specific application of the General Electric Instrument Setpoint Methodology to the Cooper Nuclear Station (CNS).

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 in selected Allowable Values for the instrumentation included in proposed Section 3.3 of the Technical Specifications is the result of application of the CNS instrumentation setpoint methodology. This methodology incorporates the guidance of ISA Recommended Practice ISA–RP67.04, Part II, ''Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation,'' September 1994. Application of this methodology results in instrumentation selected Allowable Values which more accurately reflect total instrumentation loop accuracy as well as that of test equipment and setpoint drift between Surveillances. The proposed change will not result in any hardware changes. The instrumentation included in proposed Section 3.3 of the Technical Specifications is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to this change. As a result, the proposed change will not result in unnecessary plant transients.

The role of the proposed Section 3.3 instrumentation is in mitigating and thereby limiting the consequences of accidents. The Allowable Values have been developed to ensure that the design and safety analysis limits will be satisfied. The methodology used for the development of the Allowable Values ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses and that the results and consequences described in the safety analyses remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events. 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 is the result of application of the CNS instrumentation setpoint methodology and do not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed Allowable Values does not impact safe operation of CNS in that the safety analysis limits will be maintained. The proposed Allowable Values involve no system additions or physical modifications to systems in the station.

These Allowable Values were developed using a methodology to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoints may be changed. Since operational methods remain unchanged and the operating parameters have been evaluated to maintain the station within existing design basis criteria, no different type of failure or accident is created.

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

The proposed change does not involve a reduction in a margin of safety. The proposed changes have been developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

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

NRC Project Director: John N. Hannon

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

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from the current Technical Specifications (CTS) to the improved Technical Specifications (ITS), would add an additional action statement to a limiting condition for operation (LCO). The LCO is in the Improved Standard Technical Specifications (ISTS, NUREG–1433, Revision 1) 3.6.2.3 on the residual heat removal suppression pool cooling subsystems. The requirements in the proposed ITS 3.6.2.3 on the subsystems do not exist in the CTS. The Action B for ITS 3.6.2.3 would require that if the two such subsystems were inoperable, one subsystem would have to be restored to operability within 8 hours or the plant would be in ITS 3.0.3. ITS 3.0.3 governs plant operation if an LCO (i.e., ISTS 3.6.2.3) and the associated action statement are not met (i.e., Action B to ISTS 3.6.2.3).

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:

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 (the) mitigation of an accident or transient event. The more restrictive requirements continue to ensure * * * systems, and components ((i.e., the residual heat removal suppression pool cooling subsystems)) are maintained consistent with the safety analyses and licensing basis. Therefore, this (the proposed) change does not involve a significant (an) increase in the probability or consequences of any 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, this change is consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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 increases the margin of plant safety. As provided in the discussion of the change, each change in this category (i.e., more restrictive requirements) is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements (systems and components) within the safety analyses and licensing basis. 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: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

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

NRC Project Director: John N. Hannon

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

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from the current Technical Specifications (CTS) to the improved Technical Specifications (ITS), would add an additional test (i.e., water and sediment content within limits) of diesel fuel oil that could be used in place of a current test (i.e., clear and bright appearance with proper color) in the diesel fuel oil testing program. The current tests are listed in CTS 4.9.A.2.d/e. The testing program will be in the new ITS 5.5.9. The additional test is change number 25 to Section 5.0 of the Improved Standard Technical Specifications (NUREG–1433, Revision 1).

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. (This) more stringent (requirement) do(es) not result in operation that will increase the probability of initiating an analyzed event and do(es) not alter assumptions relative to (the) mitigation of an accident or transient event. The more restrictive (requirement) continue(s) to ensure * * * systems and components (i.e., the diesel generators) are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change does not involve an increase in the probability or consequences of any 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. However, this change is consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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 increases the margin of plant safety. As provided in the discussion of the change, each change in this category (i.e., a more restrictive requirement) is, by definition, providing additional restrictions to enhance plant safety. The change maintains (systems and components) within the safety analyses and licensing basis. 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: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

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

NRC Project Director: John N. Hannon

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

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from the current Technical Specifications (TS) to improved TS for the Cooper Nuclear Station (CNS), would relocate the Trip Level Settings for the Rod Block Monitor from Table 3.2.C of the current TS to the Core Operating Limits Report. Also, details relating to the Alternate Shutdown system design and operation are proposed to be relocated from current TS 3.2.I and 4.2.I to the improved TS 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. The NRC staff has reviewed the licensee's analysis against the three criteria of 10 CFR 50.92(c), and has determined the following:

The proposed changes relocate certain details from the Technical Specifications to the Bases and the Core Operating Limits Report (COLR). The Bases and the COLR containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition, the Bases and COLR are subject to the applicable change control provisions of Chapter 5.0, Administrative Controls'', of the proposed improved Technical Specifications. Since any changes to the Bases or the COLR will be evaluated per the requirements of 10 CFR 50.59 or other applicable change control provisions, no increase in the probability or consequences of an accident previously evaluated will result. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not involve any physical alterations to the plant (no new or different type of equipment will be installed), or changes in the methods governing normal plant operation. The proposed changes will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, the details to be transposed from the TS to the Bases and the COLR are unchanged. Since any future changes to these details in the Bases or the COLR will be evaluated per the requirements of 10 CFR 50.59 or other applicable change control provisions, no reduction in a margin of safety will result. As such, these proposed changes do not involve a significant reduction in a margin of safety.

Based on the above discussion, 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 Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

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

NRC Project Director: John N. Hannon

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

Date of amendment request: April 8, 1998.

Description of amendment request: The proposed change would revise Technical Specifications (TSs) 4.4.5.3, Steam Generators—Inspection Frequencies, and 3.4.6.2.c, Reactor Coolant System (RCS) Leakage, and the associated bases to accommodate fuel cycles of up to 24 months with respect to the allowed time interval between steam generator inservice inspections.

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

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

Extending Surveillance Requirement (SR) 4.4.5.3 to accommodate a 24 month cycle for inspection of steam generator tubes structural integrity, as well as, imposing a more restrictive Limiting Condition for Operation (TS 3.4.6.2.c) for reactor coolant system leakage through Category C–2 steam generators, will neither exacerbate nor significantly increase the probability or consequences of an accident previously evaluated in the Seabrook Station [updated final safety analysis report] UFSAR.

The proposed changes to SR 4.4.5.3 do not alter the intent or method by which the surveillances are conducted, do not involve physical changes to the plant, do not alter the way structures, systems or components (SSCs) function, and do not modify the manner in which the plant is operated.

The proposed change to TS 3.4.6.2.c imposes more restrictive limits on plant operations due to RCS leakage through steam generators. The proposed change does not involve physical changes to the plant or alter the way a SSC functions.

The proposed changes to SR 4.4.5.3 and TS 3.4.6.2.c, and their associated Bases, will not adversely affect the ability of the steam generators to perform their intended safety function. Furthermore, the proposed changes do not adversely affect the physical protective boundaries of the plant. The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls or the procedural details associated with aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube eddy current test (ECT) inspections, it is concluded that the wear growth rate of the only active degradation mechanism (Anti-Vibration Bar (AVB) wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval.

Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes will not significantly increase the probability or consequences of any previously analyzed accident.

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

The proposed changes to TS 3.4.6.2 and SR 4.4.5.3, and associated Bases, do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change ( ) to the surveillance intervals for SR 4.4.5.3 is still consistent with the basis for the interval. The intent or method of performing the surveillances remains unchanged. The more restrictive limit for leakage through any one steam generator placed in Category C–2, as well as, the requirement to do an engineering assessment of steam generator tube integrity, provides additional margin of ensuring safe plant operation.

In addition, there is no adverse affect on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube ECT inspections, it is concluded that the wear growth rate of the only active degradation mechanism (AVB wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval. Therefore, extension of the current surveillance intervals to accommodate a 24 month cycle will not significantly degrade the ability, the availability or the reliability of the steam generators to perform their intended safety function, thus, it is concluded that there is no significant reduction in a margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

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

NRC Project Director: Cecil O. Thomas

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: April 6, 1998.

Description of amendment request: The proposed amendment will modify the Technical Specifications (TSs) by (1) adding a surveillance requirement to verify pressurizer heater capacity to TS 3.4.4, ''Reactor Coolant System—Pressurizer,'' (2) moving the identification of the location of the containment air temperature detectors from the surveillance requirements portion of TS 3.6.1.5, ''Containment Systems—Air Temperature,'' to the TS Bases for Containment Systems, Section 3/4.4.6.1.5, ''Air Temperature,'' and (3) modifying the action statements and surveillance requirements of TS 3.7.1.5, ''Plant Systems—Main Steam Isolation Valves.'' The TS Bases would also be updated to include the list of containment air temperature detectors and reflect the proposed changes.

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

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

The proposed change to add a surveillance requirement (SR) 4.4.4.2 to verify pressurizer heater capacity will help ensure the pressurizer will be able to function as designed to maintain Reactor Coolant System pressure. There will be no effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to modify the wording of SR 4.6.1.5 and to relocate the list of containment air temperature detectors from SR 4.6.1.5 to the Bases will not affect the Technical Specification limit for containment temperature or the frequency of verification of this limit. The proposed changes do not alter the way any structure, system, or component functions. The initial assumption for containment temperature used in the design basis accident analysis will remain the same. There will be no affect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the action statements and surveillance requirements of Technical Specification 3.7.1.5 will not affect the operability requirements of the main (steamline) isolation valves (MSIVs). There will be no effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated or on any equipment important to safety. Therefore, the License Amendment Request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences or an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes 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 proposed changes will add SR 4.4.4.2 to verify pressurizer heater capacity, relocate the list of containment temperature detectors used to verify containment temperature from SR 4.6.1.5 to the associated Bases, and modify the action statements and surveillance requirements of Technical Specification 3.7.1.5.

These changes will have no adverse effect on equipment important to safety. This equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the technical Specifications affected by these proposed changes.

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

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

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

NRC Deputy 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: April 13, 1998

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) by adding a new TS 3.5.5, ''Emergency Core Cooling Systems—Trisodium Phosphate (TSP).'' Also, the surveillance requirements in TSs 4.5.2.c.3 and 4.5.2.c.4 would be relocated to new TS 3.5.5 as TS 4.5.5.1 and TS 4.5.5.2, respectively. The applicable TS Index page and Bases sections will be updated to reflect the proposed changes.

Changes to the current requirements for the TSP are also proposed. The TSP requirements in TS 4.5.2.c.3 would become the limiting conditions for operation in the new TS; the amount of TSP required would increase from ''equal to or greater than 110 cubic feet'' to ''equal to or greater than 282 cubic feet'' based on the new calculations; the applicability would be expanded to include all of Mode 3; the action statement would allow 48 hours to restore the TSP volume; and changes would also be made to the required tests and specific details would be relocated to the applicable TS 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to relocate the current trisodium phosphate (TSP) dodecahydrate Technical Specification requirements from the surveillance requirements for the Emergency Core Cooling System to a new TSP Technical Specification will not change the requirement to store TSP inside containment. The proposed changes will require a large quantity of TSP to be stored inside containment. This large quantity, based on a recently revised calculation, will ensure sufficient TSP is available for containment sump water pH control. These proposed changes do not alter the way any structure, system, or component functions. There will be no adverse effect on any design basis accident previously evaluated, on any equipment important to safety, or o n the radiological consequences of any design basis accident. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in 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 change to increase the TSP volume stored inside containment will require two of the wire mesh TSP baskets inside containment to be replaced by two new and larger wire mesh baskets. The design of the new baskets has been evaluated and it is consistent with the requirements for equipment installed in containment. The replacement of the two wire mesh baskets will not result in any significant change in plant configuration and will not require any new or unusual operator actions. It will alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated. It will not introduce any new failure modes. Therefore, the proposed changes 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 proposed changes will relocate the current Technical Specification requirements for TSP to a new Technical Specification. The minimum required volume will be increased to reflect the results of a new calculation performed to support the current requirement to raise containment sump pH [equal to or greater than] 7.0. These changes will have no adverse effect on equipment important to safety. This equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by these proposed changes.

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: April 11, 1997 (supersedes July 26, 1996, application)

Description of amendment request: The proposed amendment would modify the Monticello Technical Specifications (TS) sections 3.6.C, Coolant Chemistry, and 3/4.17.B, Control Room Emergency Filtration System. The changes were proposed to establish TS requirements consistent with modified analysis inputs used for the evaluation of the radiological consequences of the main steam line break accident.

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

A limit is established in the plant Technical Specifications for steady state radioiodine concentration in the reactor coolant to ensure that in the event of a release of radioactive material to the environment due to a postulated high energy line break up to and including a design basis Main Steam Line Break Accident, radiation doses are maintained within the guidelines of 10 CFR part 100. The steady state radioiodine concentration in the reactor coolant is an input for analysis of the radiological consequences of an accident due to a Main Steam Line Break outside of containment and postulated high energy line breaks. In addition, requirements are established in the Technical Specifications for control room habitability. During an accident, the control room emergency filtration system provides filtered air to pressurize the Control Room to minimize the activity, and therefore the radiological dose, inside the control room.

A change is proposed for the steady state radioiodine concentration. This value is conservative with respect to the value used in the Main Steam Line Break dose consequences analysis and is consistent with the dose consequences evaluation of a postulated Reactor Water Cleanup (RWCU) line break. Changes are proposed to the limiting conditions for operation and surveillance requirements for the Control Room Emergency Filtration Train iodine removal efficiency. These changes are consistent with the inputs used in the analysis of the radiological consequences of the postulated RWCU line break and the Main Steam Line Break Accident. These proposed requirements maintain operating restrictions for analytical inputs used in the analysis of the Main Steam Line Break Accident. Evaluation of these events has demonstrated that the postulated radiological consequences will remain within the licensing basis established in the AEC [Atomic Energy Commission] Provisional Operating License Safety Evaluation Report, dated March 18, 1970, thus the proposed changes do not result in an increase in the consequences of previously evaluated accidents.

The analysis of the Main Steam Line Break Accident performed using a reactor coolant radioiodine concentration of 2 (microcuries)/gm dose equivalent Iodine-131 and a control room ventilation filter efficiency consistent with the proposed Technical Specifications changes demonstrated that radiological consequences of the Main Steam Line Break are not changed significantly. The radiological consequences of the Main Steam Line Break Accident remain within the exposure guidelines of 10 CFR part 100 and 10 CFR part 50 appendix A, General Design Criterion 19. The offsite dose consequences remain bounded by the licensing basis provided in the AEC Provisional Operating License Safety Evaluation Report, dated March 18, 1970. The control room doses calculated for the hot standby Main Steam Line Break Accident using the TID–14844 dose conversion factors remain bounded by the dose consequences of the comparable design basis loss of coolant accident.

The evaluation of the postulated RWCU line break, performed using a reactor coolant radioiodine concentration of 0.25 (microcurie)/gm dose equivalent Iodine-131 and a control room ventilation filter efficiency consistent with the proposed Technical Specifications changes, demonstrated that the radiological consequences of this event remain within the exposure guidelines of 10 CFR part 100 and 10 CFR part 50 Appendix A, General Design Criterion 19. The offsite dose consequences remain bounded by the Main Steam Line Break as established in the licensing basis provided in the AEC Provisional Operating License Safety Evaluation Report, dated March 18, 1970.

The proposed Technical Specification changes do not introduce new equipment operating modes, nor do the proposed changes alter existing system inter-relationships. The proposed changes do not introduce new failure modes. The system improvements to reduce bypass leakage during postulated accidents do not have an adverse effect on control room habitability. Therefore, this amendment will not cause a significant increase in the probability of an accident previously evaluated for the Monticello plant.

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

The proposed Technical Specification changes do not introduce new equipment operating modes, nor do the proposed changes alter existing system inter-relationships. Operator action to mitigate the consequences of the postulated RWCU line break is conservative based on the very limited action required by the operator to close the containment isolation valves and the availability of control room indications to alert the operator to the postulated break. The use of a ten (10) minute operator response time to take manual actions in response to postulated events is consistent with Monticello's licensing basis for similar events. The use of operator actions and all available equipment is consistent with current regulatory guidance for mitigating the consequences of postulated line breaks.

The proposed change to the specification for reactor coolant dose equivalent radioiodine is conservative with respect to the re-evaluation of the Main Steam Line Break Accident for the more conservative hot standby initial condition for the postulated accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is consistent with the postulated high energy line break of a Reactor Water Cleanup line. The proposed changes to the limiting conditions for operation and surveillance requirements for the control room emergency filtration train iodine removal efficiency are consistent with the inputs used in the evaluation of the radiological consequences of the postulated RWCU line break and the Main Steam Line Break Accident. The system improvements to reduce bypass leakage during postulated accidents do not have an adverse effect on control room habitability. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident.

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

Surveillance data has demonstrated the proposed requirements are within the current capability of the facility. The proposed changes maintain margins of safety. These proposed requirements maintain operating restrictions for analytical inputs used in the analysis of the bounding postulated high energy line break of a Reactor Water Cleanup line and the Main Steam Line Break Accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is conservative with respect to the re-evaluation of the Main Steam Line Break Accident for the more conservative hot standby initial condition for the postulated accident. The proposed change to the specification for reactor coolant dose equivalent radioiodine is consistent with the postulated high energy line break of a Reactor Water Cleanup line. The evaluation of these postulated events determined that the radiological consequences remain within the exposure guidelines of 10 CFR part 100 and of 10 CFR part 50 Appendix A, General Design Criterion 19. The proposed changes to the limiting conditions for operation and surveillance requirements for the control room emergency filtration train iodine removal efficiency provide assurance that the system will perform at the filter efficiency as used in the evaluation of the radiological consequences of the postulated events. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter

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

Date of amendment request: April 10, 1998.

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 to revise TS 6.2.2.g and 6.3 to change the name of the Operations Manager to Operations Director and to change the requirement for the Operations Director to hold a senior reactor operator (SRO) license.

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 to revise the title of the Operations Manager to Operations Director is an administrative change that clarifies the Technical Specification (TS) to reflect current position titles.

The proposed change provides assurance that the Operations Director will continue to have knowledge of pressurized water reactor (PWR) operation and emergency event mitigation. The proposed change does not detract from the Operations Director's ability to perform his primary responsibilities. In this case, by having previously held a senior reactor operator (SRO) license, the Operations Director has achieved the necessary training, skills, and experience to fully understand the operation of plant equipment and the watch requirements for operators. In summary, the proposed change does not affect the ability of the Operations Director to provide the plant oversight required of his position.

Additionally, another off-shift individual that holds an SRO license for Diablo Canyon Power Plant (DCPP) directs the licensed activities of licensed operators (an Operations middle manager) will have specific knowledge of operation and emergency event mitigation at DCPP. This will assure that the change in qualification of the Operations Director does not affect the probability of an operator initiating an accident or increasing the consequences of an accident due to improper direction from management. The training and qualification programs for operators on shift will not be affected by the proposed changes.

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

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

The proposed change to revise the title of the Operations Manager to Operations Director is an administrative change that clarifies the TS to reflect current position titles.

The proposed change to TS 6.2.2g. and 6.3 do not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. It does not affect the performance of NRC licensed operators since the proposed changes do not impact the training or qualification of any operator on shift. Operation of the plant in conformance with TS and other license requirements will continue to be supervised by personnel who hold an SRO license. The proposed change to TS 6.2.2g and 6.3 ensures that the Operations Director will be a knowledgeable and qualified individual by requiring the individual to have held an SRO license at a PWR.

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

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

The proposed change to revise the title of the Operations Manager to Operations Director is an administrative change that clarifies the TS to reflect current position titles.

The proposed change involves an administrative control that is not related to the margin of safety. The proposed change does not reduce the level of knowledge or experience required of an individual who fills the Operations Director position, nor does it affect the conservative manner in which the plant is operated. The on-shift licensed operators will continue to be supervised by personnel who hold an SRO license in accordance with 10 CFR 50.54(l).

Therefore, neither of the proposed changes involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: March 26, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.8.2.1, ''AC Distribution—Operating,'' to add operability conditions and action statements for the 115-volt vital instrument bus (VIB) D and inverter. The proposed amendments complete the recommended action from NRC Generic Letter 91–11, Resolution of Generic Issues 48, ''LCOs for Class 1E Vital Instrument Buses,'' and 49, ''Interlocks and LCOs for Class 1E Tie Breakers'' pursuant to 10 CFR 50.54(f), dated July 18, 1991.

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, as described above, does not make any physical changes to the plant or components, nor changes the manner in which the plant or components are operated as a result of the addition of the Note and the D VIB and Inverter to the TS. The proposed change incorporates the operating requirements of the Technical Specification Interpretation (TSI) developed in response to GL 91–11 into the Salem Unit 1 and 2 Technical Specifications. Incorporating this interpretation into the Technical Specifications eliminates the need for the TSI.

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

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

The proposed change does not introduce any design or physical configuration change to the plants, change the function of the 115 Volt D VIBs and inverters, or the manner in which they are maintained or tested.

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

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

The proposed Action Times associated with the incorporation of the D VIB into the Technical Specifications are consistent with the current Action Times for the A, B, and C VIBs for a loss of an AC bus. Adding the note to the Salem Unit 1 Technical Specification brings consistency between Salem Units 1 and 2, and is also consistent with NUREG 1431, Vol. 1, Rev 1 ''Standard Technical Specifications Westinghouse Plants.''

The outage duration limit of 72 hours for the D inverter is acceptable based on the following: (1) the proposed 72 hours Action Time to restore the inoperable inverter to operable is supported by a PSA [probabilistic safety assessment] assessment. NRC Draft SRP [Standard Review Plan] Chapter 16.1, Revision 13, ''Risk-Informed Decision making: Technical Specifications'' notes that an incremental conditional core damage probability (ICCDP) of 5.0 E–7 is considered very small. The proposed 72 hour allowable outage time was calculated utilizing the NRC incremental conditional core damage probability (ICCDP), and (2) the inoperability of the D VIB Inverter will not affect the operation of any Safeguard Equipment Cabinet (SEC) or Emergency Diesel Generator (EDG).

Therefore, the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038

NRC Project Director: Robert A. Capra.

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

Date of amendment request: April 18, 1997, as supplemented by letters dated October 10, 1997, and February 27, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.7.6, ''Plant Systems—Control Room Emergency Ventilation System.'' Additional Limiting Conditions for Operation would be added related to the availability of the station vent normal range radiation monitoring instrumentation. The associated TS bases would also be modified consistent with these changes. The staff's proposed no significant hazards consideration determination for the requested change was published on June 4, 1997 (62 FR 30646).

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes 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 no accident initiators, conditions, or assumptions are affected by the proposed changes.

The proposed change to Limiting Condition for Operation (LCO) 3.7.6.1 would include new required Action statements in the event that one or both channels of Station Vent Normal Range Radiation Monitoring instrumentation become inoperable. Under the proposed Action statements for inoperable Station Vent Normal Range Radiation Monitoring instrumentation, should the control room normal ventilation system be isolated and at least one train of the control room emergency ventilation system be placed in operation, these systems would be in a state equivalent to that which they would be in following an actual high radiation condition. These proposed changes have no bearing on the probability of an accident.

The proposed change to the terminology utilized in Surveillance Requirement (SR) 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes have no bearing on the probability of an accident.

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

As described above, under the proposed Action statements for inoperable Station Vent Normal Range Radiation Monitoring instrumentation, should the control room normal ventilation system be isolated and at least one train of the control room emergency ventilation system be placed in operation, these systems would be in a state equivalent to that which they would be in following an actual high radiation condition. Therefore, in the unlikely event of an accident requiring control room isolation while in this condition, the dose consequences to control room operators would be unchanged.

The proposed change to the terminology utilized in Surveillance Requirement (SR) 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes have no bearing on the consequences of an accident.

2. Not create the possibility of a new or different kind of accident from any accident

previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.

As described above, under the proposed Action statements for inoperable Station Vent Normal Range Radiation Monitoring instrumentation, should the control room normal ventilation system be isolated and at least one train of the control room emergency ventilation system be placed in operation, these systems would be in a state equivalent to that which they would be in following an actual high radiation condition. Operation of the equipment and components in this manner would not introduce the possibility of any new or different kinds of accidents.

The proposed change to the terminology utilized in Surveillance Requirement (SR) 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes would not introduce the possibility of any new or different kinds of accidents.

3. Not involve a significant reduction in a margin of safety because the proposed changes to the Action under LCO 3.7.6.1 ensure that control room isolation capability is maintained in the event a station vent radiation monitor is inoperable. The proposed allowable outage time of seven days for one inoperable channel is consistent with the presently allowable outage time for one inoperable CREVS. The proposed Action to place at least one CREVS train in operation within one hour, in the event both channels of radiation monitoring become inoperable, is more conservative than the present Action which requires that a plant shutdown commence within one hour, but does not require the CREVS be placed in operation.

The proposed change to the terminology utilized in Surveillance Requirement (SR) 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes would not affect 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, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606

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

NRC Acting Project Director: Richard P. Savio

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

Date of application request: March 9, 1998.

Description of amendment request: The proposed amendment application would revise Technical Specification 3/4.5.2b.1 and its associated Bases to add clarification in regard to venting the emergency core cooling system (ECCS) pump casings and accessible discharge piping high points. Technical Specification 3/4.5.2b.1 requires verification that the ECCS piping is full of water at least once per 31 days by venting the ECCS pump casings, i.e., the safety injection pump, residual heat removal pump, and centrifugal charging pump casings and accessible discharge piping high points. The centrifugal charging pump (CCP) casings do not have installed casing vents. Instead of a casing vent, the suction and discharge piping is installed as vertical runs attached to the top-mounted suction and discharge nozzles of each CCP pump. Information provided by the pump manufacturer indicates that the vertical configuration of the piping is sufficient to prevent the accumulation of noncondensible gases that could cause gas binding. Therefore the CCP casings are effectively vented by vents on the CCP discharge lines. The proposed amendment application would revise Technical Specification 3/4.5.2b.1 and associated Bases to require the residual heat removal and safety injection pump casings and accessible ECCS discharge piping high points be vented to ensure the ECCS piping is full of water.

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

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

The proposed change will align the surveillance requirements with the installed system design and normal operating conditions. The performance of surveillances required by Technical Specifications is not postulated to initiate an accident. The intent of the surveillance ensures OPERABILITY of the ECCS by verifying that the ECCS piping is full of water and not subjected to gas binding or water hammer. The design of the CCPs is such that significant noncondensible gases do not collect in the pumps, whether they are running or not. Therefore, it is unnecessary to require periodic pump casing venting to ensure the CCPs will remain OPERABLE. In addition, operating experience has shown that no significant voiding has occurred in the affected piping which will continue to be vented at a high point every 31 days per Surveillance Requirement 4.5.2b.1). Therefore, no increase in the probability or consequences of an accident will occur as a result of this change.

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

The proposed change will not result in new failure modes because there are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The design of the CCPs is such that significant noncondensible gases do not collect in the pumps, whether they are running or not. Therefore, it is not necessary to require periodic pump casing venting to ensure the equipment will remain OPERABLE. Manual venting operations will be performed to minimize the potential for voids in system piping. Accordingly, this change will not create the possibility of a new or different kind of accident.

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protective functions. There will be no impact on any margin of safety.

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

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149

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

NRC Project Director: William H. Bateman

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: December 18, 1997.

Description of amendment request: The proposed changes revise the Technical Specifications (TS) to clarify the terminology used to describe equipment surveillances performed with a refueling interval frequency. Currently the TS are somewhat ambiguous in the wording in this regard, and the proposed changes would adhere to the improved Standard TS and make it clear whether the reactor must be shutdown when performing the test, or whether a ''refueling interval'' frequency (e.g., 18 months) is intended. All of the clarifications are in Section 4 of the TS. In addition, minor typographical errors are being corrected, and an obsolete reference is proposed to be deleted.

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—Operation of Surry Units 1 and 2 in accordance with the proposed Technical Specifications change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an accident is not increased as a result of the proposed Technical Specification change since surveillance intervals are being clarified, not changed, and will continue to validate system/component availability, operability and performance during the appropriate unit mode. The proposed change is administrative in nature, therefore, station operations are not being affected. The consequences of an accident previously evaluated are not increased since station operations are not being changed, and no physical modifications are being made to plant systems or components.

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

As noted above, the proposed change is administrative in nature. A new or different type of accident is not being created since no new accident precursors are being introduced and equipment surveillances will continue to be performed as required to ensure proper system/component operation. Plant systems are not being modified, system operations are not being affected, and equipment surveillance intervals are not being increased. Consequently, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The proposed Technical Specifications change does not involve a significant reduction in a margin of safety.

This is an administrative change. Clarification of refueling surveillance interval terminology to ensure consistency in application does not affect plant equipment performance. Surveillance intervals are not being increased, and equipment surveillance tests performed on a refueling interval frequency (i.e. once per 18 months) will continue to ensure system/component performance as assumed in the existing safety analyses. Therefore, the proposed Technical Specification change does not involve a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of § 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: P.T. Kuo, Acting

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: March 25, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) Sections 6.1.A; 6.1.A.2; 6.1.C.1.a and b; 6.1.C.1.f.1,4 and 8; 6.1.C.1.g.1 and 3; 6.8.A.2; and 6.8.B.2 for Units 1 and 2, changing the title of Station Manager to Site Vice President, and the titles of the Assistant Station Managers to Manager-Station Operations and Maintenance and Manager-Station Safety and Licensing.

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:

Virginia Electric and Power Company has reviewed the proposed Technical Specifications changes against the criteria of 10 CFR 50.92 and has concluded that the changes do not pose a significant hazards consideration. Specifically, station operations in accordance with the proposed Technical Specifications changes will not:

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

The proposed changes are administrative in nature. The overall responsibility for safe operation and review of plant operations is not being changed. There are no changes to the operation of any plant system or its design as a result of these changes. Therefore, neither the probability of occurrence nor the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report are increased.

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

The proposed changes are administrative in nature. The overall responsibility for safe operation and review of plant operations is not being changed. There are no changes to the operation of any plant system or its design that could create any new modes of operation or accident precursors. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

c. The proposed changes do not result in a significant reduction in margin of safety as defined in the basis for any Technical Specifications.

The proposed changes are administrative in nature. The overall responsibility for safe operation and review is not being changed. There are no changes to the operation of any plant system or its design as a result of these changes. Safety systems are maintained operable as required by Technical Specifications. Therefore, the margin of safety is not changed.

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: P.T. Kuo, Acting

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

Date of amendment request: April 8, 1998.

Description of amendment request: The change would reduce allowable reactor coolant system (RCS) specific activity from 1.0 microcurie/gram to 0.35 microcurie/gram dose equivalent I–131.

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

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

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

The change implements a more restrictive RCS activity limit. Specific RCS activity is an initial plant condition and, therefore, is not an accident initiator and can not cause the occurrence of or increase the probability of an accident. The change also lowers the curve of Figure TS 3.1–3 which restricts operation with high specific activity. The new value for specific activity is justified by the Westinghouse calculation which demonstrates acceptable offsite and control room doses following a (main steamline break) MSLB with a maximum allowable primary to secondary leak rate. By lowering the RCS specific activity and maintaining leakage within the projected maximum allowable, 10 CFR 100 and GDC 19 criteria are satisfied. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the RCS specific activity limit will not significantly effect operation of the plant nor will it alter the configuration of the plant. There will be no additional challenges to the main steam system or the reactor coolant system pressure boundary and no new failure modes are introduced. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Reduction of the RCS specific activity limit allows an increase in the MSLB allowable primary to secondary leakage. The net effect is no reduction in the margin of safety provided by 10 CFR part 100 and GDC 19 criteria. The maximum allowable leakage is the leakage limit for projected SG leakage following SG tube inspection and repair. Reducing specific activity to increase projected leak rate follows guidance given by GL 95–05 and effectively takes margin available in the specific activity limits and applies it to the projected SG leak rate. This has been determined to be an acceptable means for accepting higher projected leak rates while still meeting the applicable limits of 10 CFR part 100 and GDC 19 criteria with respect to offsite and control room doses. Additionally, monitoring of the specific activity and compliance with the required actions remains unchanged. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

For consistency, the value of secondary coolant activity in Table TS 4.1.2 is being corrected from 1.0 microcurie/gram to 0.1 microcurie/gram. This is consistent with a previously submitted and approved amendment, therefore, no significant hazards exist for 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: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497

NRC Project Director: Richard P. Savio

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

Date of amendment request: April 15, 1998.

Description of amendment request: The revisions in the proposed Technical Specification amendment are part of the licensee's fuel and reload change plan for Cycle 23. The revisions implement changes associated with a new fuel design and also reflect changing plant conditions due to steam generator tube plugging and repair. The Technical Specifications (TS) would be modified as follows:

(1) Figure 2.1–1 would be revised to reflect the recently approved High Thermal Performance (HTP) Critical Heat Flux (CHF) correlation and corresponding Departure from Nucleate Boiling Ratio (DNBR) limit of 1.14. The figure would also reflect changes in peak rod power and minimum reactor coolant flow.

(2) TS 3.10.b—new hot channel factors would be incorporated for the new fuel design and the corresponding increase in peaking factors. The limits for Height Dependent Nuclear flux Hot Channel Factor are specified in TS 3.10.b.1 and the limits for Nuclear Enthalpy Rise Hot Channel Factor are specified in 3.10.b.2.

(3) TS 3.10.k—the specification for the maximum Reactor Coolant System (RCS) Inlet Temperature would be replaced with a specification for the maximum Reactor Coolant System (RCS) Average Temperature.

(4) TS 3.10.l—the statement ''During 100% steady-state power operation'' would be revised in the specification for minimum Reactor Coolant System (RCS) pressure and replaced with ''During steady-state power operation.''

(5) TS 3.10.m—the minimum Reactor Coolant Flow is being decreased to 85,500 gallons per minute per loop.

(6) TS 3.10.n—would be revised to reflect the new Minimum DNBR limit.

(7) Figure TS 3.10–1—the Required Shutdown Reactivity vs. Boron Concentration would be revised to reflect the change to an 18 month fuel cycle.

(8) Figure TS 3.10–2, the Hot Channel Factor Normalized Operating Envelope would be revised to reflect the values used in the new safety analyses.

(9) The Table of Contents and the Basis sections would be revised to accommodate the above 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:

Figure TS 2.1–1: The proposed changes will not:

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

The safety limits curves are not accident initiators. Therefore, the change will not increase the probability of an accident previously evaluated. The proposed changes to the safety limits curves do not alter the plant configuration, operating set points, or overall plant performance. The safety limits curves reflect the changes to the DNBR limit, CHF correlation, RCS flow peaking factors and fuel design. The significant hazards determinations for these parameters are evaluated later in this submittal. Therefore, the change will not increase 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 changes in the safety limits curves do not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

Operation in the acceptable regions (i.e., below and to the left of the safety limit curves) in combination with the reactor protection and engineered safety systems designed into the plant will ensure that the safety limits are not exceeded during normal operation or during anticipated design basis operational transients. The core will be operated in the nucleate boiling heat transfer regime. Departure from nucleate boiling (DNB) will not occur and therefore fuel cladding integrity will be assured.

The revised safety limit curves have been developed using operating parameters at their bounding values (e.g., rod powers at the peaking factor limits, reactor coolant flow at the minimum operating limit). The revised curves will bound plant operation with Siemens Power Corporation standard or heavy fuel. Therefore, this change will not involve a significant reduction in safety margin.

TS 3.10.b: The proposed changes will not:

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

Peaking factor limits are input assumptions to the safety analyses and are not accident initiators. Therefore, this change would not increase the probability of occurrence of an accident previously evaluated.

The safety analyses input assumptions are designed to bound actual plant operation. Changing the safety analysis input assumption for the increased peaking factor limits does not change the underlying progression of design basis accidents evaluated in the safety analyses. All safety analysis acceptance criteria are satisfied in the increased peaking factor limit conditions. Additionally, the radiological consequences are bounded by existing analysis at the increased peaking factor limits. Therefore, this change will not significantly increase the consequences of an accident previously analyzed.

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

This change incorporates the safety analyses assumptions for core peaking factor limits for Siemens Power Corporation heavy fuel. The change does not alter plant equipment, set points or plant performance. Therefore, changing the peaking factor limits for analysis purposes will not create a new or different kind of accident from any accident previously evaluated.

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

Results of the safety analyses and of radiological consequences indicate that all acceptance criteria are satisfied. The peaking factor limits assumed in the safety analyses are consistent with the proposed revised limits and these revised limits are established to bound actual plant operation. Therefore, this change will not involve a significant reduction in the margin of safety.

TS 3.10.k: The proposed change will not:

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

The RCS average temperature limit is not an accident initiator. Changing the technical specification limit consistent with the accident analyses will not increase the probability of an accident previously evaluated.

The proposed change limits the maximum reactor coolant system average temperature to 568.8 °F. The design basis safety analyses, the Large and Small Break LOCA accidents and the non-LOCA accidents, have been analyzed and/or evaluated consistent with the revised RCS average temperature. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified temperature. Therefore, the change will not increase the consequences of an accident previously evaluated.

The proposed technical specification limit for maximum allowed RCS average temperature was decreased below the analytical limit to account for instrument error.

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

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the revised reactor coolant system average temperature. The TS limit will bound actual plant operation. Therefore, there is no significant reduction in the margin of safety.

TS 3.10.l: The proposed change will not:

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

The RCS pressure limit is not an accident initiator. By removing the 100% value from the specification, the assumptions in the safety analyses are not changed. Changing the technical specification to remove the 100% power criteria will not increase the probability of an accident previously evaluated.

The design basis safety analyses have been analyzed and/or evaluated at the specified RCS pressure. The analyses and evaluations have demonstrated that all safety analyses acceptance criteria are satisfied at this pressure. Therefore, the change would not increase the consequences of an accident previously evaluated.

The proposed technical specification limit for minimum allowed RCS pressure was increased above the analytical limit to account for instrument error.

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

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the reactor coolant system pressure. The limit will bound actual plant operation. Therefore, there is no significant reduction in the margin of safety.

TS 3.10.m: The proposed change will not:

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

The RCS flow limit is not an accident initiator. Changing the technical specification limit consistent with the accident analysis will not increase the probability of an accident previously evaluated.

The proposed change limits the minimum reactor coolant flow. The design basis safety analyses have been analyzed and/or evaluated at the revised RCS flow. The re-analysis and evaluation have demonstrated that all safety analysis acceptance criteria are satisfied at the specified flow. Therefore, the change will not significantly increase the consequences of an accident previously evaluated.

The proposed technical specification limit for minimum allowed RCS flow was increased above the analytical limit to account for instrument error.

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

The proposed change does not alter the plant configuration or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change is consistent with the safety analyses. All safety analyses acceptance criteria are satisfied at the revised reactor coolant system flow. The limit will bound actual plant operation.

The change reduces the RCS flow rate limit. Re-analysis of LOCA and non-LOCA transients determined all safety requirements of KNPP accident analyses were still met at the reduced RCS flow rate limit. Therefore, this proposed change does not significantly reduce the margin of safety.

TS 3.10.n: The proposed change will not:

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

The Departure from Nucleate Boiling Ratio (DNBR) is not an accident initiator. Therefore, the change in the DNBR will not increase the probability of an accident previously evaluated.

The proposed change to the DNBR value does not change plant configuration, operating set points, or overall plant performance. Therefore, the change will not increase 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 change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

All safety analyses acceptance criteria are satisfied using the HTP CHF correlation. The DNBR limits assumed in the safety analyses will bound actual plant operation and assures at 95/95 that DNBR will not occur. Therefore, there is no reduction in the margin of safety.

TS Figure 3.10–1: The proposed change will not:

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

Required Shutdown Reactivity vs. Boron Concentration was revised to reflect the longer cycle length and the resulting increase in boron concentration. The Required Shutdown Reactivity vs. Boron Concentration is not an accident initiator. Extending the boron concentrations to account for longer fuel cycles will not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change is consistent with the cycle length and core physics analyses for longer fuel cycles. Operation within the limits specified in the figure will assure all core safety evaluation acceptance criteria are satisfied. The limit will bound actual plant operation. Therefore, there is no reduction in the margin of safety.

TS Figure 3.10–2: The proposed change will not:

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

The Hot Channel Factor Normalized Operating Envelope figure was revised to reflect the values used in the safety analyses.

The Hot Channel Factor Normalized Operating Envelope figure is not an accident initiator. Changing the technical specification figure consistent with the assumptions of the accident analyses will not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the plant configuration, operating set points, or overall plant performance. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change is consistent with the safety analyses. Operation within the limits specified in the figure will assure all safety analyses acceptance criteria are satisfied. The limit will bound actual plant operation. 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: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497

NRC Project Director: Richard P. Savio

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

Date of amendment request: May 2, 1995, October 12, 1995, March 26, 1996, and December 15, 1997 (TSCR 172)

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) Table 15.4.1–1, ''Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels,'' to change the test frequencies for radiation monitors as discussed in Generic Letter 93–05 (''Line-Item Technical Specifications Improvements To Reduce Surveillance Requirements For Testing During Power Operation''), remove the radiation monitoring system as item 36, revise note(s), and add those radiation monitors and their surveillance requirements that support current TS or meet the requirements of 10 CFR 50.36. Additionally, several typographical and nomenclature errors would be corrected. This amendment request was initially noticed in the **Federal Register** on June 6, 1995 (60 FR 29890).

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:

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

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents. Initiating events for accidents previously evaluated for the Point Beach Nuclear Plant (PBNP) include control rod withdrawal and drop, chemical volume control system malfunction (boron dilution), startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, loss of all alternating current (ac) power to the auxiliaries, turbine overspeed, fuel handling accidents, accidental releases of waste liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures.

These proposed changes do not cause an increase in the probabilities of any accidents previously evaluated because these changes will not cause an increase in the probability of any initiating events for accidents previously evaluated. In particular, these changes affect the radiation monitoring system surveillance requirements and make administrative changes that will not result in changing accident initiators.

The consequences of the accidents previously evaluated in the Final Safety Analysis Report (FSAR) are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems.

The proposed changes reduce the burden associated with radiation monitoring system required surveillance by establishing surveillances for only the necessary monitors (i.e., elimination of the testing requirement for monitors that do not perform a required function) and changing the testing frequency for these monitors from monthly to quarterly. The proposed changes do not increase the probability of failure of this equipment or its ability to operate as required for the accidents previously evaluated in the PBNP FSAR. The proposed changes to correct typographical errors and correct nomenclature are administrative only and do not increase the probability of an accident previously evaluated nor do they affect the consequences of any accident previously evaluated.

Therefore, these proposed license amendments do not affect the consequences of any accident previously evaluated in the PBNP FSAR because the factors that are used to determine consequences of accidents are not being changed.

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

New or different kinds of accidents can only be created by new or different accident initiators or sequences. The changes proposed by this license amendment request do not create any new or different accident initiators or sequences because the revisions to TS Table 15.4.1–1, ''Minimum Frequencies for Checks, Calibrations, and Tests of Instrument Channels,'' will not cause failures of equipment or accident sequences different than the accidents previously evaluated. The proposed changes to correct typographical errors and correct nomenclature are administrative only. Therefore, these proposed TS changes do not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR.

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

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection. The changes proposed by this license amendment request provide the appropriate surveillance requirements for the radiation monitoring system. The revised surveillance requirements will continue to ensure that the required radiation monitors will operate as required. The design and operation of the reactor and containment are not affected by these proposed changes. The proposed changes to correct typographical errors and correct nomenclature are administrative only. Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed.

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 considerations.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Cynthia A. Carpenter

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.

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: February 23, 1998, as supplemented March 27, 1998.

Brief description of amendment: The proposed amendment would allow addition of a footnote to the Safety Limit Minimum Critical Power Ratio value in the Technical Specifications and the associated action statement.

Date of publication of individual notice in the **Federal Register**: April 10, 1998 (63 FR 17900).

Expiration date of individual notice: May 11, 1998.

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

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

Date of amendment request: April 3, 1998, and related application dated November 22, 1995, as supplemented February 19, April 19, May 3, June 12, and December 4, 1996, and January 30 and August 7, 1997.

Description of amendment request: The proposed amendment would revise Technical Specification 3.8.1.1 to change the emergency diesel generator allowed outage time from 3 to 7 days. This would be a one-time amendment, effective from the date of issuance until September 30, 1998.

Date of publication of individual notice in **Federal Register**: April 13, 1998 (63 FR 18048).

Expiration date of individual notice: May 13, 1998.

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

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

Date of amendment request: April 9, 1998, TXX–98107.

Description of amendment request: The proposed amendment would allow on a one time basis, the verification of the proper operation of the Unit 2 load shed seal-in contacts and the diesel generator trip bypass contacts at power and crediting performance of Surveillance Requirements (SR) 4.8.1.1.2f.4(a) and 4.8.1.1.2f.6(a), at power as opposed to ''during shutdown'' as currently required by those SR. The proposed amendment would also allow on a one time basis the verification of the proper operation of the Unit 2 lockout relays and contacts to be deferred until the startup from 2RFO4 or earlier outage to at least MODE 3.

Date of individual notice in the **Federal Register**: April 20, 1998.

Expiration date of individual notice: May 5, 1998.

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

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 Ch. 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.

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

Date of application for amendment: March 17, 1997, as supplemented April 13, 1998. The April 13, 1998, submittal contained clarifying information only, and did not change the proposed no significant hazards consideration.

Brief description of amendment: The amendment revises Technical Specifications 4.1.2.2.c, 4.5.2.e, 4.6.2.1.c, 4.6.2.2.c, 4.6.3.2, 4.7.1.2.1.b, 4.7.3.b, and 4.7.4.b to delete specific restrictions in the text of the surveillances that the tests must be done while the unit is shut down.

Date of issuance: April 14, 1998.

Effective date: April 14, 1998

Amendment No.: 77.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19826)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 14, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 12, 1997.

Brief description of amendments: The amendments modify the bypass logic for Main Steam Line Isolation Valve Isolation Actuation Instrumentation on Condenser Low Vacuum as stated in Technical Specification Tables 3.3.2–1 and 4.3.2–1.

Date of issuance: April 14, 1998.

Effective date: Immediately, to be implemented prior to startup from L1F35 for Unit 1 and from L2R07 for Unit 2.

Amendment Nos.: 124 and 109.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6982).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 14, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 18, 1997, as supplemented by letter dated January 26, 1998.

Brief description of amendments: The amendments revise the operating license of Unit 1 and Unit 2 to (1) delete license conditions that have been fulfilled; (2) delete exemptions that have expired; (3) update information to reflect current plant status and regulatory requirements; and (4) make other corrections and editorial changes.

Date of issuance: April 23, 1998.

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

Amendment Nos.: Unit 1–164; Unit 2–156.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6983).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 3, 1998.

Brief description of amendments: The amendments revise the Technical Specifications to change the qualification requirements for the members of the Safety Review Group.

Date of issuance: April 27, 1998.

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

Amendment Nos.: Unit 1–165; Unit 2–157.

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

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14486).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application of amendments: August 28, 1997. Supplement January 22, February 19, March 19, and April 6, 13, and 17, 1998.

Brief description of amendments: The amendments incorporate new testing and operability requirements related to the installation of new systems and upgrades associated with the Emergency Condenser Circulating Water System. Review of the system for this amendment also includes a review of the new design features incorporated into the upgrade and its acceptability as a safety grade system.

Date of Issuance: April 24, 1998.

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

Amendment Nos.: Unit 1–229; Unit 2–230; Unit 3–226

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

Date of initial notice in **Federal Register**: September 24, 1997 (62 FR 50002).

The January 22, 1998, February 19, March 19, and April 6, 13, and 17, 1998, letters provided clarifying information that did not change the scope of the August 28, 1997, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 16, 1998.

Brief description of amendments: These amendments add a new Limiting Condition for Operation (LCO) 3.0.6 to TS Section 3/4.0, ''APPLICABILITY.'' The new LCO 3.0.6 provides specific guidance for returning equipment to service under administrative control to perform testing required to demonstrate OPERABILITY.

Date of issuance: April 15, 1998.

Effective date: Both units, effective immediately, to be implemented within 30 days.

Amendment Nos.: 213 and 90.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (63 FR 14142, March 24, 1998). That 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 April 23, 1998, 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 determination of no significant hazards consideration are contained in a Safety Evaluation dated April 15, 1998.

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

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: June 26, 1997, as supplemented by letter dated September 11, 1997.

Brief description of amendment: The amendment changes the Appendix A TSs by modifying Tables 3.7–1 and 3.7–2. The revision to Table 3.7–1 changes the Main Steam Safety Valves (MSSVs) orifice size from 26 square inches to 28.27 square inches and relocates the orifice size from the TS Table to the TS Bases. The change to correct the orifice size is an editorial change to make the TS consistent with plant design. The changes to Table 3.7–2 delete the provisions that allows continued plant operation with three MSSVs inoperable. The proposed amendment will also revise TS Bases 3/4.7.1.1 to remove the equation used for determining the reduced maximum allowable linear power level-high reactor trip settings of TS Table 3.7–2.

Date of issuance: April 20, 1998.

Effective date: April 20, 1998, to be implemented within 30 days.

Amendment No.: 142.

Facility Operating License No. NPF–38: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 16, 1997 (62 FR 38135).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1998.

No significant hazards consideration comments received: No.

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

GPU Nuclear, Inc. and Saxton Nuclear Experimental Corporation (SNEC), Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF)

Date of application for amendment: November 25, 1996, as supplemented on May 30, June 4 and 16, August 21 and September 16, 1997, and February 3 and 9, 1998, and March 31, 1998. During the amendment request review, the staff also referred to the SNEF Decommissioning Environmental Report dated April 17, 1996, licensee responses to NRC questions about the environmental report dated July 18, 1996, and March 3 and 31, 1998, the SNEC Facility Updated Safety Analysis Report, Revision 0, submitted on October 25, 1996, Revision 1, submitted on August 21, 1997, and Revision 2, submitted on February 3, 1998, and the SNEC Facility Decommissioning Quality Assurance Plan submitted by letter dated November 8, 1996, as supplemented on May 30, 1997, and February 3 and 9, 1998.

Brief description of amendment: The amendment allows decommissioning of the SNEF. The changes to the license and Technical Specifications (TSs) (1) accommodate decommissioning activities at the SNEF, (2) establish specific TS controls over decommissioning activities, (3) establish limiting conditions for performing decommissioning activities, (4) extend exclusion area controls to include the SNEF Decommissioning Support Facility, (5) establish requirements for a Radiological Environmental Monitoring Program, and an Offsite Dose Calculation Manual, and (6) establish requirements for technical and independent safety reviews. In addition, the amendment authorizes other administrative and editorial changes to the TSs associated with the changes described above.

Date of issuance: April 20, 1998.

Effective date: April 20, 1998.

Amendment No.: 15.

Amended Facility License No. DPR–4: Amendment changed the Amended Facility License and TSs.

Date of initial notice in **Federal Register:** March 12, 1997 (62 FR 11494).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Saxton Community Library, Front Street, Saxton, Pennsylvania 16678

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

Date of application for amendment: December 16, 1996, as supplemented September 11, 1997 and March 25, 1998.

Brief description of amendment: The amendment (1) reflects the change in the legal name of the operator of TMI–1 from GPU Nuclear Corporation to GPU Nuclear, Inc., and (2) reflects in the TMI–1 Facility Operating License the registered trade name of GPU Energy now used by the owners of the facility.

Date of Issuance: April 24, 1998.

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

Amendment No.: 207.

Facility Operating License No. NPF–50: Amendment revised the Facility Operating License and the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4350).

The September 11, 1997 and March 25, 1998, submittals provided clarifying information and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 7, 1997.

Brief description of amendment: The amendment revised the Technical Specifications surveillance requirements to change setpoints for the refueling platform main hoist overload cutoff, loaded interlock, and redundant loaded interlock due to planned modifications to the refueling platform mast.

Date of issuance: April 16, 1998.

Effective date: As of the date of issuance to be implemented upon completion and acceptance of design modifications to the refueling platform mast.

Amendment No.: 81.

Facility Operating License No. NMF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** December 31, 1997 (62 FR 68309).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 13, 1998, as supplemented March 25, 1998.

Brief description of amendment: The amendment modifies the Technical Specification requirements associated with the Minimum Critical Power Ratio (MCPR) safety limits for Cycle 19 based on the cycle-specific analysis of the current mixed core of GE [General Electric] 11, GE10, four GE12 lead use assemblies, and eight SPC [Siemens Power Corporation] ATRIUM–9B assemblies.

Date of issuance: April 20, 1998.

Effective date: April 20, 1998.

Amendment No.: 100.

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

Date of initial notice in **Federal Register:** March 20, 1998 (63 FR 13704).

The March 25, 1998, letter provided clarifying information in response to the staff's request for additional information during a teleconference. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. Therefore, renoticing was not warranted.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1998.

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

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of application for amendment: October 14, 1997, as supplemented on March 26, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.4.6.3, ''Primary Coolant System Pressure Isolation Valves Limiting Condition for Operation,'' to add additional pressure isolation valves, establish the operability and testing requirements for the pressure isolation valves, and make this section more consistent with Salem Unit 2 TSs.

Date of issuance: April 20, 1998.

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No.: 210.

Facility Operating License No. DPR–70: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* November 19, 1997 (62 FR 61845).

The March 26, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1998.

No significant hazards consideration comments received: No.

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

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: January 26, 1998.

Brief description of amendments: The proposed amendments would (1) modify the requirement to hold a Susquehanna Steam Electric Station (SSES) Senior Reactor Operator (SRO) license in Section 6.3.1 for the Manager-Nuclear Operations (MNO), (2) replace the position of MNO with Operations Supervisor—Nuclear in the Section 6.2.2g requirement to hold an SSES SRO license and (3) renumber existing TS Section 6.3.1 to include 6.3.1.1, 6.3.1.2, and 6.3.1.3.

Date of issuance: April 10, 1998.

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

Amendment Nos.: 175 and 147.

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

Date of initial notice in **Federal Register***:* February 24, 1998 (63 FR 9270).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 11, 1996, as supplemented March 16, 1998.

Brief description of amendment: This amendment changes the TSs to preclude the need to enter into Limiting Condition for Operation 3.0.3 to allow performance of certain emergency diesel generator testing.

Date of issuance: April 10, 1998.

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

Amendment No.: 148.

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

Date of initial notice in **Federal Register***:* March 13, 1996 (61 FR 10397).

The February 15, 1996, letter corrected the no significant hazards (NSH) determination. The NSH determination was used in the March 13, 1996 (61 FR 10397) notice. The March 24, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1998.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: January 12, 1998.

Brief description of amendment: This amendment revises TS Table 4.4.6.1.3–1 to change the withdrawal schedule for the first capsule to be withdrawn from 10 Effective Full Power Years (EFPY) to 15 EFPY. In addition, TS Surveillance Requirement 4.4.6.1.4 will be revised to remove the references to flux wire removal and analysis that was originally required following the first cycle of operation and replaced with a new surveillance requirement. The new requirement refers to the flux wires that are located within the surveillance capsules, which will be removed and analyzed in accordance with the surveillance capsule removal schedule located in Table 4.4.6.1.3–1.

Date of issuance: April 15, 1998.

Effective date: As of the date of issuance.

Amendment No.: 126.

Facility Operating License No. NPF–39: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* February 11, 1998 (63 FR 6988).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1998.

No significant hazards consideration comments received: No.

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

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: February 27, 1998.

Brief description of amendment: The amendment changes the Technical Specifications by revising the pressure-temperature curves to extend heatup and cooldown limits from 11 to 13.3 effective full-power years, provides the corresponding overpressure protection system limits, and makes some minor changes to ensure specification clarity and conservatism.

Date of issuance: April 10, 1998.

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

Amendment No.: 179.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* March 9, 1998 (63 FR 11456).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610

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

Date of application for amendment: February 26, 1998, as supplemented by letter dated March 20, 1998.

Brief description of amendment: This amendment revises Technical Specification (TS) Section 3/4.4.5, ''Reactor Coolant System—Steam Generators,'' TS Section 3/4.4.6.2, ''Reactor Coolant System—Operational Leakage,'' and the associated bases to allow use of the ''repair roll'' steam generator tube repair process.

Date of issuance: April 14, 1998.

Effective date: April 14, 1998.

Amendment No.: 220.

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

Date of initial notice in **Federal Register***:* March 9, 1998 (63 FR 11460).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 14, 1998.

No significant hazards consideration comments received: No. The supplemental information submitted by the licensees did not affect the proposed no significant hazards consideration determination.

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

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

Date of application for amendment: June 24, 1997.

Brief description of amendment: This amendment revises TS Section 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' TS Section 3/4.6.1.7, ''Containment Ventilation System,'' TS Section 3/4.6.3.1, ''Containment Isolation Valves,'' and TS Section 3/4.9.4, ''Refueling Operations—Containment Penetrations,'' and the associated TS Bases. Valve position requirements have been added, and certain containment radiation monitor requirements, valve isolation verification requirements, and containment radiation monitor optional uses have been deleted. Administrative changes have also been made.

Date of issuance: April 15, 1998.

Effective date: April 15, 1998.

Amendment No.: 221.

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

Date of initial notice in **Federal Register**: July 30, 1997 (62 FR 40858).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment requests: February 25, 1998, (TXX–98050) as supplemented by letter dated March 9, 1998, (TXX–98066) for License Amendment Request (LAR) 98–002, March 12, 1998, (TXX–98076) for LAR 98–003, and March 18, 1998, (TXX–98079) for LAR 98–004.

Brief description of amendments: This amendment is the result of three Notice of Enforcement Discretions (NOEDs) dated February 24, March 13, and 17, 1998. These NOEDs although distinct actions changed the same page of the CPSES TS therefore the single amendment is being issued to cover the three parts of this amendment.

The first part of the amendment would be a temporary change to the TSs to remove the requirement to demonstrate the load shedding feature of MCC XEB4–3 as part of Surveillance Requirements (SRs) 4.8.1.1.2f.4)a) and 4.8.1.1.2f.6)a) until the plant startup subsequent to the next refueling outage for Unit or until an outage of 24 hour in duration.

The second part of the amendment would provide a temporary Technical Specification change for SRs 4.8.1.1.2f.4)b) and 4.8.1.1.2f.6)b) to allow the verification of the auto connected shut-down loads through the load sequencer to be performed at power for fuel cycle 6 on Unit 1 and fuel cycle 4 on Unit 2.

The third part of the amendment would allow on a one time basis, crediting performance of Surveillance Requirements (SR) 4.8.1.1.2f.4)a) and 4.8.1.1.2f.6)a), during POWER OPERATIONS as opposed to ''during shutdown.'' Note that the bus tie breaker for MCC XEB4–3 for Unit 2 was not tested during the last surveillance test and was the subject of part one of this amendment.

Date of issuance: April 20, 1998.

Effective date: April 20, 1998.

Amendment Nos.: Unit 1—Amendment No. 58; Unit 2—Amendment No. 44.

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

Date of initial notice in **Federal Register**: March 9, 1998 (63 FR 11458), March 27, 1998 (63 FR 14974) and April 2, 1998 (63 FR 16287).

The Commission's related evaluation of the amendment, finding of exigent circumstances and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 20, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 31, 1997, as supplemented by letter dated February 27, 1998.

Brief description of amendment: The amendment revises the Callaway Plant, Unit 1 Technical Specifications to change setpoint and allowable stress values of certain reactor trip system (RTS) and engineered safety features actuation system (ESFAS) functional units.

Date of issuance: April 13, 1998.

Effective date: April 13, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 125.

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

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2283).

The February 27, 1998, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149

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

Date of application for amendment: December 11, 1997, as supplemented on March 3, 1998.

Brief description of amendment: The amendment revises the values for the safety limit minimum critical power ratio for Cycle 20 operation.

Date of Issuance: April 10, 1998.

Effective date: April 10, 1998, to be implemented within 30 days.

Amendment No.: 159.

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

Date of initial notice in **Federal Register**: February 11, 1998, (63 FR 7000).

The March 3,1998 supplement did not change the original proposed no significant hazards consideration.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 10, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 11, 1996, as supplemented by letter dated December 8, 1997.

Brief description of amendment: The amendment involves a change to the safety and relief valve setpoint tolerance and power operation with an inoperable safety relief valve.

Date of Issuance: April 15, 1998.

Effective date: April 15, 1998, to be implemented within 30 days.

Amendment No.: 160.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17241).

The information provided in the December 8, 1997, submittal did not change the original proposed no significant hazards determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 15, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 26, 1996.

Brief description of amendments: The proposed action would revise the Technical Specifications (TS) to eliminate the records retention requirements from Section 6.10 of the TS since these requirements have already been relocated to the Operational Quality Assurance program, Chapter 17, in revision 32 of the Updated Final Safety Analysis Report.

Date of issuance: April 13, 1998.

Effective date: April 13, 1998.

Amendment Nos.: 208 and 189.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 132).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 13, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 3, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Surveillance Requirement Tables 3.3–1 and 4.3–1 for both units, modifying the testing requirements for the reactor trip bypass breaker.

Date of issuance: April 14, 1998.

Effective date: April 14, 1998.

Amendment Nos.: 209 and 190.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 11, 1998 (63 FR 11925).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 14, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 18, 1997.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Surveillance Requirements 4.7.1.7.2.a.1 and 4.7.1.7.2.a.2 for both units, modifying the testing frequency of the Turbine throttle and Governor valves.

Date of issuance: April 16, 1998.

Effective date: April 16, 1998.

Amendment Nos.: 210 and 191.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66146)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 16, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 3, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Surveillance Requirement 4.4.10.1.1, modifying the inspection requirements for the Reactor Coolant Pump (RCP) flywheels for both units and eliminating the examination requirements for the flow straighteners in each steam generator to the RCP elbow on Unit 1.

Date of issuance: April 22, 1998.

Effective date: April 22, 1998.

Amendment Nos.: 211 and 192.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 11, 1998 (63 FR 11924)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 22, 1998.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Md., this 29th day of April 1998.

For the Nuclear Regulatory Commission.

Stuart A. Richards,

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

[FR Doc. 98–11911 Filed 5–5–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application to Withdraw from Listing and Registration; (Oryx Technology Corp., Common Stock, $0.001 Par Value; Common Stock Warrants) File No. 1–12680

April 30, 1998.

Oryx Technology Corp. (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified securities (''Securities'') from listing and registration on the Pacific Exchange, Inc. (''PCX'' or ''Exchange'').

The reasons cited in the application for withdrawing the Securities from listing and registration include the following:

The Securities of the Company have been listed for trading on the Exchange and, pursuant to a Registration Statement of Form 8–A, effective on April 5, 1994, the National Association of Securities Dealers Automated Quotation System (''NASDAQ''). Trading in the Company's Securities on the NASDAQ commenced at the opening of business on April 6, 1994, and concurrently therewith on the PCX.

The Company has complied with Exchange Rule 3.4(b) by filing with the Exchange a certified copy of the resolutions adopted by the Company's Board of Directors authorizing the withdrawal of the Securities from listing and registration on the PCX and by setting forth in detail to the Exchange the reasons for and facts supporting the proposed delisting. In deciding to withdraw its Securities from listing and registration of the PCX, the Company considered the direct and indirect costs and expenses attendant on maintaining the dual listing of its Securities on the NASDAQ and the PCX. The Company does not see any particular advantage in the dual trading of its Securities and believes that dual listing will fragment the market for its Securities.

By letter, the Exchange informed the Company that it has no objection to the withdrawal of the Company's Securities from listing and registration on the PCX.

By reason of Section 12 of the Act and the rules and regulations thereunder, the Company shall continue to be obligated to file reports under Section 13 of the Act with the Commission.

Any interested person may, on or before May 21, 1998, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, N.W., Washington, D.C. 20549, facts bearing upon whether the application has been made in accordance with the rules of the Exchange and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.

Jonathan G. Katz,

Secretary.

[FR Doc. 98–11988 Filed 5–5–98; 8:45 am]

BILLING CODE 8010–01–M

SECURITIES AND EXCHANGE COMMISSION

[File No. 500–1]

Solucorp Industries, Ltd.; Order of Suspension of Trading

April 30, 1998

It appears to the Securities and Exchange Commission that there is a lack of current and accurate information concerning the securities of Solucorp Industries Ltd. (''Solucorp'') because of questions regarding the accuracy of assertions by Solucorp in documents sent to and statements made to market makers of the stock of Solucorp, other broker dealers, and to investors concerning, among other things: (1) the negotiation, existence and terms of contracts entered into by Solucorp during the period July 1, 1995 through the present; (2) revenues purportedly accrued under a license agreement with Smart International Ltd. and reported in financial statements for the quarter ended September 30, 1997 and the six-month period ended December 31, 1997, which were included in a registration statement and transition report filed with the Commission in December 1997 and April 1998, respectively; and (3) revenues projected in press releases on August 27, 1997, October 24, 1997 and April 16, 1998.

The Commission is of the opinion that the public interest and the protection of investors require a suspension of trading in the securities of the above listed company.

Therefore, it is ordered, pursuant to Section 12(k) of the Securities Exchange Act of 1934, that trading in the above listed company is suspended for the period from 9:30 a.m. EST, May 1, 1998 through 11:59 p.m. EST, on May 14, 1998.

By the Commission.

Jonathan G. Katz,

Secretary.

[FR Doc. 98–12060 Filed 5–1–98; 3:53 pm]

BILLING CODE 8010–01–M