pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''International Uranium (USA) Corporation Envirocare's Appeals of LBP–99–11 and LBP–99–20 (Environcare's Dismissal for Lack of Standing)'' (PUBLIC MEETING) be held on July 7, and on less than one week's notice to the public.

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

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: July 9, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–18052 Filed 7–12–99; 1:08 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from June 19, 1999, through July 2, 1999. The last biweekly notice was published on June 30, 1999 (64 FR 35199).

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 August 13, 1999, 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 No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: June 15, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications to incorporate the performance-based 10 CFR 50 Appendix J, Option B for Type A tests (containment integrated leakage rate tests). Option B will be implemented for Type A testing in accordance with NRC Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' dated September 1995, and Nuclear Energy Institute (NEI) Guideline 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' dated July 26, 1995. Type B and C testing (containment penetration leakage tests) will continue to be performed in accordance with 10 CFR 50 Appendix J, Option A.

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

The Harris Nuclear Plant (HNP) Type A testing history provides justification for the proposed test schedule change to one test in a 10 year period. With the successful Type A tests of September 1992 and May 1997, and a greater than 24 month elapsed time between the two tests, CP&L considers the requirement of two consecutive Type A tests to have been met. This testing has affirmed the acceptable reliability of the containment structure to minimize leakage as designed, and provides assurance that its performance to continuously function as designed is not challenged due to this test schedule extension to once in 10 years.

This proposed change to revise the test schedule frequency does not impact or alter the design of any system, structure or component. The limit on allowable leakage is not increased. Type A testing provides periodic verification of the leak tight integrity of the containment and the components that penetrate the containment structure. NUREG–1493, Section 10.1.2, ''Leakage-Testing Intervals,'' states that reducing the frequency of Type A tests from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk.

Therefore, based on these considerations, and the previous plant-specific Type A test results, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes only incorporate the performance-based testing approach authorized in 10 CFR 50 Appendix J, Option B, and are justified based on previous plant-specific Type A test results. Plant structures, systems, and components will not be operated in a different manner as a result of these proposed changes and no physical modifications to equipment are involved. The interval extensions allowed by Option B of 10 CFR 50 Appendix J do not have the potential for creating the possibility of a new or different type of accident from any previously evaluated.

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

The proposed changes do not change the allowable leak rate from the containment; they only allow an extension of the interval between the performance of Type A leak rate testing. NUREG–1493 provides the technical basis for the NRC's rulemaking to revise containment leakage testing requirements for nuclear power reactors in 10 CFR 50 Appendix J. NUREG–1493, Section 10.1.2, ''Leakage-Testing Intervals,'' states that increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk.

Based on these considerations and the previous plant-specific Type A test results, the proposed changes do not involve a reduction in the margin of safety.

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

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

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

NRC Section Chief: Sheri R. Peterson.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: June 15, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification 4.7.D.6 by replacing the leakage limit of 11.5 standard cubic feet per hour (scfh) for each main steam isolation valve (MSIV) with a limit of 46 scfh on the total combined leakage from all four main steam lines.

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 changes to the Technical Specifications, Appendix A, modifies the allowed MSIV leakage limit to an aggregate value with no change to the total allowed leakage rate. This change does not affect either the automatic or manual features that would close the MSIVs. Performance of the leakage tests do not adversely affect any accident previously evaluated. Consequently, this proposed amendment 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 safety function of the MSIVs is to provide a timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. The MSIVs are designed to automatically close whenever plant conditions warrant main steam line isolation. Changing the leakage limits to include an aggregate value does not affect the isolation function. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The total allowed leakage rate for all MSIVs remains unchanged at 46 scfh. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite, and, thus, the radiological analyses remain unchanged and within the guidelines of 10 CFR 100 and General Design Criteria 19. Therefore, these changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 19, 1999.

Description of amendment request: The proposed amendments would relocate Technical Specification Section 3/4.4.4, ''Chemistry'' from the TS to the Updated Final Safety Analysis Report and Administrative Technical Requirements.

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

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

The proposed changes simplify the TS, meet regulatory requirements for relocated TS, and implement the recommendations of the NRC's Final Policy Statement on TS improvements. The Chemistry requirements will be relocated to the Updated Final Safety Analysis Report (UFSAR) and Administrative Technical Requirement that has been incorporated into the UFSAR by reference. Future changes to these requirements will be controlled by 10 CFR 50.59. The proposed changes are administrative in nature and do not involve any modification to any plant equipment or affect plant operation. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any previously evaluated accident.

Consequently, this proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety related system performs its function. Therefore, this proposed TS amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment represents the relocation of current requirements that are based on generic guidance or previously approved provisions for other stations. The proposed changes are administrative in nature and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Duane Arnold Energy Center and Quad Cities Nuclear Power Station. Since the proposed changes are administrative in nature, and are based on NRC accepted provisions which have been adopted at other nuclear facilities, and maintain the necessary levels of system reliability, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: June 3, 1999.

Description of amendment request: The proposed amendment would delete sections of the Technical Specifications that no longer apply to the Haddam

Neck Plant's permanently shutdown and defueled condition; increase the weight of loads allowed over the spent fuel pool; relocate certain definitions and requirements from the Technical Specifications to the Technical Requirements Manual (TRM), Connecticut Yankee Quality Assurance Program (CYQAP), or the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM); and correct typographical errors, renumber sections, and repaginate the Technical Specifications.

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

The proposed changes do not involve an SHC [significant hazards consideration] because the changes would not:

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

In the present plant configuration, the reactor-related accidents previously evaluated (i.e., LOCA [loss-of-coolant accident], MSLB [main steamline break], etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases. The following events are presently considered as bounding of all other events:

—Fuel handling and cask drop accidents in the spent fuel building,
—Criticality in the spent fuel pool,
—Loss of spent fuel cooling,
—Resin fire (gaseous release), and
—Rupture of a tank containing radioactive liquid.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased, with a corresponding decrease in the frequency of fuel movement. The radiological consequences of a fuel handling accident, should one occur, decrease the longer the spent fuel is allowed to decay. As discussed previously, the spent fuel inventory of radioactive iodine and noble gases [, with the exception of Kr–85,] have decayed more than 20 half-lives since shutdown and are no longer a release concern. With this reduced source the results of the fuel handling accident show that the filters of Specification 3.9.12 are no longer necessary. The allowed weight over the spent fuel pool is still less than that previously [evaluated]. Therefore, there has been no increase in the probability or consequences of a fuel handling or cask drop accident.

Criticality controls are imposed by specifications ¾.9.13 and ¾.9.14 * * * [The requirements of these specifications have not been changed.] Therefore, there has been no increase in the probability or consequences of a criticality event.

Spent fuel cooling is maintained by keeping the pool temperature below 150°F. Should normal cooling be lost, the availability of an abundant supply of water ensures that sufficient time is available prior to boiling to restore cooling. This is controlled by specifications ¾.9.11 and ¾.9.16 * * * [The requirements of these specifications have not been changed. Technical specification ¾.9.15 does not apply to the permanently defueled condition of the plant. Therefore, there has been no increase in the probability or consequences of a loss of cooling event.]

The probability of a gaseous or liquid radioactive release is not changed by the proposed revisions. As the plant undergoes decommissioning, the previous limiting events [such as a loss-of-coolant-accident] are no longer applicable, and previous non-limiting events [such as a resin fire] now become limiting. These * * * events have not changed from how they might have occurred in the past. The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident during defueled operation and a spent resin fire during processing of resin from the reactor coolant system decontamination. The rupture of a tank containing radioactive liquid was assessed and found to be bounded by these events. With the plant defueled and permanently shutdown, the demands on the radwaste systems are lessened since no new radioisotopes are being generated by irradiation or fission. Therefore, there is no increase in the probability or consequences of a gaseous or liquid radioactive release.

* * * * *

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

The proposed changes are generally of an administrative nature and do not have an effect on the physical plant. The events considered bound other potential events and are considered the limiting cases for potential gaseous or liquid releases to the environment.

With the plant undergoing decommissioning, the types of accidents one might be concerned with involve criticality of the spent fuel, or draining of the spent fuel pool. None of the proposed changes affect the possibility of such an event. Also, none of the proposed changes could lead to a radiological release of a greater magnitude than for the events considered, such as might occur with the accumulation of a greater quantity of radioactive material in one location, or with damage to a greater number of fuel assemblies than considered in the fuel handling accident.

The proposed changes restrict the operations that can be conducted at the plant, and do not permit any new type of activity from what had previously been authorized. The effect on systems, structures and components affected by the proposed changes have no adverse impact on the storage of fuel nor on the processing of radioactive wastes presently at the site. The present set of limiting events are a subset of events previously considered. Therefore these changes do not create the possibility of a new or different kind of accident from any accident previously considered.

* * * * *

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

The proposed changes have no impact on the analyses of postulated design basis events remaining applicable to the Haddam Neck Plant. Analysis of the limiting events show that their consequences to the public are within the limits of 10 CFR [Part] 20 and the EPA PAGs [Environmental Protection Agency Protective Action Guides]. The consequences to members of the operating staff are within the limits of 10 CFR [Part] 50, Appendix A, General Design Criterion 19, ''Control Room Habitability''. Therefore there is no reduction in a margin of safety.

* * * * *

Based on the above, the proposed changes to the operating license and technical specifications do not involve a reduction in the margin of safety due to the reduced decay heat load, the decay of radionuclides since shutdown, and by maintaining the heavy load restriction.

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

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

Attorney for licensee: Mr. J. A. Ritsher, Ropes & Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Section Chief: Michael T. Masnik.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: June 10, 1999.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to adopt a Ventilation Filter Testing Program in TS Section 6.0, ''Administrative Controls,'' and remove the specific ventilation filter surveillance requirements from TS 3/4.6.4.4, ''Hydrogen Purge System,'' TS 3/4.6.5.1, ''Shield Building Emergency Ventilation System,'' and TS 3/4.7.6.1, ''Control Room Emergency Ventilation System.''

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 Number 1, in accordance with this change would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. The replacement of the specific Technical Specification (TS) ventilation filter testing Surveillance Requirements for the Containment Hydrogen Purge System 3/4.6.4.4), Shield Building Emergency Ventilation System (3/4.6.5.1), and the Control Room Emergency Ventilation System (3/4.7.6.1), with a reference to the newly created Ventilation Filter Testing Program contained in TS Administrative Controls Section 6.8.4.f, Ventilation Filter Testing Program, is a removal and relocation of certain TS details. The proposed TS 6.8.4.f will, however, add controls to maintain similar operation, maintenance, testing and system operability for these three ventilation systems. The TS Bases changes reflect the use of the Ventilation Filter Testing Program. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. No physical alterations of the DBNPS are involved, nor are plant operating methods being changed. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated. No new or different types of failures or accident initiators are being introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because no inputs into the calculation of any Technical Specification Safety Limit, Limiting Safety System Settings, Technical Specification Limiting Condition for Operation, or other previously defined margins for any structure, system, or component important to safety are being affected by the 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: 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 Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 10, 1999.

Description of amendment request: The proposed amendment would correct the regulation referenced in Section 5.8, ''High Radiation Area,'' of the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS). The ITS currently references 10 CFR 20, Paragraph 20.1601(2) and (3), whereas the correct reference is 10 CFR 20.1601(c).

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 the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) is editorial in nature. The change involves revising the incorrect reference in ITS Section 5.8.1 to the correct Code of Federal Regulations reference that pertains to controlling access to high radiation areas. The proposed ITS change does not involve any change to plant design, operation, maintenance, or procedures. As a result, no changes to the plant are being made which would impact either the contributors to an accident or to the consequences of an accident.

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

The proposed editorial change to the ITS does not involve any changes to any plant structure, system, or component (SSC) or to its operation or maintenance. There is no impact on any equipment that would be considered as contributors to either new or different accidents. Thus, the change to the ITS does not create the possibility of a new or different kind of accident.

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

The proposed change to the ITS involves a reference change and does not involve the design or operation of any plant SSC. No changes to the methods for controlling access to high radiation areas are proposed. No changes to the methods for controlling personnel and/or activities in high radiation areas are proposed. Therefore, the proposed 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 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–A 5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: May 17, 1999.

Description of amendment request: The proposed amendment would revise a note in Surveillance Requirement (SR) 3.3.8.1 in the Crystal River Unit 3 Improved Technical Specifications (ITS). The note currently states that, when Emergency Diesel Generator (EDG) Loss Of Power Start instrumentation is placed in an inoperable status solely for performance of this surveillance, entry into associated Conditions and Required Actions may be delayed for up to four hours provided the two channels monitoring the Function for the bus are OPERABLE or tripped. The proposed revision to the note states that entry into the Conditions and Required Actions of ITS Section 3.3.8 is not required provided the applicable Conditions and Required Actions of ITS Section 3.8.1, ''Electrical Power Systems, AC Sources—Operating,'' are entered for the EDG being made inoperable. The proposed amendment would also delete a superceded 60-day surveillance frequency and the associated note which indicated that the frequency was not effective after November 23, 1997.

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?

Proposed Improved Technical Specifications (ITS) Change A—Revision of Surveillance Note

The note in Surveillance Requirement (SR) 3.3.8.1 involves the timing for placing Crystal River Unit 3 (CR–3) into the applicable Conditions and Required Actions when performing the surveillance. The proposed revision will make the note consistent with the actual method of performing the surveillance at CR–3. The design and testing configuration does not allow CR–3 to use the relief provided by the note. As a result, the Conditions and Required Actions of ITS Section 3.3.8 are entered at the start of the surveillance performance. The design and testing configuration requires entry into ITS Section 3.8.1 Conditions prior to performing SR 3.3.8.1. The revised note would require entering the applicable Conditions and Required Actions of ITS Section 3.8.1 for one Emergency Diesel Generator (EDG) being inoperable. This approach and the proposed note are conservative relative to the current note. The proposed note results in no changes to the method or to the timing of performing SR 3.3.8.1. Direct entry into ITS Section 3.8.1 Conditions will achieve the same final ITS condition as if Section 3.3.8 Conditions and Required Actions were entered. Therefore, the probability of occurrence and the consequences of any accident previously evaluated are unaffected by this change.

Proposed ITS Change B—Deletion of Frequency Note

The note under Frequency in SR 3.3.8.1 involves the period of time that the 60 day surveillance frequency would be in effect. The 60 day frequency was a temporary extension that was needed to implement modifications to the EDG during the 1997 CR–3 design outage. This was a one-time extension of the frequency. The note indicates this temporary nature of the 60 day frequency. Deleting the note is an editorial change since the surveillance has reverted back to its 31 day frequency and the note is no longer effective. Because the proposed deletion of the note is an editorial change, and no change is proposed to the current 31 day frequency, the probability of occurrence and the consequences of any accident previously evaluated are unaffected by this change.

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

Proposed ITS Change A—Revision of Surveillance Note

The proposed revision of the note in SR 3.3.8.1 involves only the timing of entry into associated ITS Conditions and Required Actions. No changes are proposed to the existing ITS Conditions and Required Actions. The proposed change is conservative since it will require entering the appropriate Conditions and Required Actions immediately upon starting SR 3.3.8.1. Changing the timing for entry into ITS Conditions and Required Actions does not create the possibility of a new or different kind of accident from those evaluated previously.

Proposed ITS Change B—Deletion of Frequency Note

Deletion of the note under SR 3.3.8.1 Frequency is an editorial change since the note is no longer effective. The current frequency for performing SR 3.3.8.1 is 31 days. This is the same frequency that was in effect prior to the one-time, temporary change of the frequency to 60 days. The editorial change of deleting the note that is no longer effective does not create the possibility of a new or different kind of accident from those evaluated previously.

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

Proposed ITS Change A—Revision of Surveillance Note

One manner in which a margin of safety related to a Surveillance Requirement might be affected would be if entry into a Limiting Condition for Operation (LCO) were delayed. The result of a delay in entering an LCO would be an increase in the time before a Required Action was taken, such as commencing a plant shutdown. Generally, such allowed times reflected in ITS Required Actions are based on some margin. Increasing the time allowed before starting a certain Required Action might result in a reduction of a margin of safety. However, the proposed ITS change allows entry into Section 3.8.1 Conditions immediately rather than after a delay. The proposed ITS change does not change the final plant condition required by the ITS. Therefore, the proposed ITS change does not result in a reduction in a margin of safety.

Proposed ITS Change B—Deletion of Frequency Note

Another manner in which a margin of safety related to a surveillance requirement might be affected would be if the frequency of performance were changed. Generally, margin might be reduced if the frequency were reduced (i.e., the interval between performing surveillances were increased). This proposed editorial change to delete the note in SR 3.3.8.1 Frequency does not result in a change to the surveillance frequency. Thus, the proposed deletion of the note does not affect the existing 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: 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 Section Chief: Sheri R. Peterson.

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

Date of amendment request: June 11, 1999.

Description of amendment request: The proposed amendment makes various plant organization title 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. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR [Updated Final Safety Analysis Report] transient analyses. No Technical Specification Limiting Condition for Operation, Action statement or Surveillance Requirement is affected by any of the proposed changes. These proposed changes do not reduce the level of qualification, authority or accountability associated with the affected Technical Specification responsibilities. Further, the proposed changes do not alter the design, function, or operation of any plant component. Therefore, the proposed amendment does not affect the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or modes of plant operation defined in the plant operating license, or Technical Specifications that preserve safety analysis assumptions. The proposed changes do not introduce a new mode of plant operation or surveillance requirement, nor involve a physical modification to the plant. The proposed changes do not alter the design, function, or operation of any plant components. Therefore, the proposed amendment does not affect the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a reduction in a margin of safety. The proposed changes are administrative in nature. There is no reduction in the organization position qualifications, authority and accountability associated with the affected Technical Specification responsibilities. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the UFSAR transient analyses. No Technical Specification Limiting Condition for Operation, Action statement, or Surveillance Requirement is affected. Therefore, the proposed amendment does not reduce the margin of safety.

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

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

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

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: April 12, 1999.

Description of amendment request: The proposed amendment would revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.7 to allow a representative sample of reactor instrumentation line excess flow control valves (EFCV) to be tested every 24 months, instead of testing each EFCV every 24 months.

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 current SR frequency requires each reactor instrumentation line EFCV to be tested every 24 months. The EFCVs at DAEC are designed so that they will not close accidentally during normal operation, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and have their status indicated in the control room (reference DAEC UFSAR [updated final safety analysis report] 1.8.11). This proposed change allows a reduced number of EFCVs to be tested every 24 months. There are no physical plant modifications associated with this change. Industry operating experience demonstrates a high reliability of these valves. Neither EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore there can be no increase in the probability of occurrence of an accident regarding this proposed change.

Instrument lines connecting to the Reactor Coolant Pressure Boundary (RCPB) with EFCVs installed also have a flow-restricting orifice upstream of the EFCV. The consequences of an unisolable rupture of such an instrument line [have] been previously evaluated in response to Regulatory Guide (RG) 1.11 (DAEC UFSAR 1.8.1.1). That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (3.5 hours). Therefore, although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by this previous evaluation. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the consequences of an accident previously evaluated as a result of this change.

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

This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience demonstrates the high reliability of these valves. The potential failure of an EFCV to isolate by the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line rupture. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

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

The consequences of an unisolable rupture of an instrument line [have] been previously evaluated in response to RG 1.11 (reference DAEC UFSAR 1.8.1.1). That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (3.5 hours). The only margin of safety applicable to this proposed change is considered to be that implied by this evaluation. Since a continuous discharge was assumed in this evaluation, any potential failure of an EFCV to isolate postulated by this reduced testing frequency is bounded and 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: 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

NRC Section Chief: Claudia M. Craig

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

Date of amendment request: April 30, 1999.

Description of amendment request: The proposed amendment would revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to revise the safety function lift setpoint tolerance limits for the main safety valves (SVs) and the safety/relief valves (SRVs). The current tolerance bands for the SVs and SRVs would be revised from −3% to +1% to a new band of plus or minus 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. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change allows an increase in the as-found SV and SRVs safety mode setpoint tolerance, determined by test after the valves have been removed from service, from +1%/−3% to plus or minus 3%.

The proposed change does not alter the TS requirements on the nominal SV or SRV safety mode lift setpoints, the SRV relief mode setpoints, the required frequency for the SV or SRV lift setpoint tests, or the number of SVs and SRVs required to be operable.

Consistent with current requirements, this change continues to require that these valves be adjusted to within plus or minus 1% of their nominal lift setpoints following testing. This change does not change the behavior and operation of any SV or SRV and therefore has no significant impact on reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Updated Final Safety Analysis Report (UFSAR).

This change does not involve physical changes to the valves, nor does it change the operating characteristics or safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components. Therefore these changes will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' and were reviewed and approved by the NRC in a Safety Evaluation (SE) dated March 8, 1993. The plant specific evaluations, required by the NRC's SE and performed to support this proposed change, show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a plus or minus 3% setpoint tolerance. They also show that operation of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems will not be adversely affected and the containment response from a loss of coolant accident will be acceptable. The plant systems associated with these proposed changes will still be capable of meeting all applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the UFSAR. Therefore, these changes will not involve a significant increase in the consequences of any accident previously evaluated.

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

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

These proposed changes were developed in accordance with the provisions contained in the NRC SE, dated March 8, 1993, for the ''BWR Owners Group Inservice Pressure Relief Technical Specification Revision Licensing Topical Report,'' NEDC–31753P. The revised SV and SRV setpoint tolerance limit will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve no significant hardware changes, no significant changes to the operation of any systems or components, and no changes to existing structures, systems, or components, there can be no impact on the occurrence of any accident.

The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to allow an increase in the SV and SRV safety mode setpoint tolerance from +1%/−3% to plus or minus 3% does not alter the nominal SV or SRV lift setpoints or the number of SVs or SRVs required to be operable. This change does not involve physical changes to the valves, nor does it change the operating characteristics or the safety function of the valves. The proposed change does not involve a physical alteration of the plant. No new or different equipment is being installed. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

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

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

The proposed change does not involve a significant reduction in a margin of safety. Establishment of the plus or minus 3% SV and SRV setpoint tolerance limit will not adversely impact the operation of any safety related component or equipment. Engineering evaluations concluded that there are no significant impacts on fuel thermal limits, safety related systems, structures or components, and no significant impact on the accident analyses associated with the proposed changes.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results.

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.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: May 10, 1999.

Description of amendment request: The proposed amendment would revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Section 2.1.1.2, to revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) to support operation with GE–12 fuel with a 10x10 pin array.

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.

There is no change to any plant equipment other than the fuel. The SLMCPR protects the fuel in accordance with the design bases. The SLMCPR calculations limit the bundle power to ensure the critical power ratio remains unchanged. Therefore, there is not an increase in the probability of transition boiling. The basis of the SLMCPR calculation remains the same, ensuring that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. Therefore, there is no increase in the probability of occurrence of a previously evaluated accident.

The fundamental sequences of accidents have not been altered. The Minimum Critical Power Ratio (MCPR) Operating Limits are selected such that potentially limiting accidents do not cause the MCPR to decrease below the SLMCPR anytime during the accident. Therefore, there is no impact on any of the limiting accidents. Therefore there is no increase in the consequences of any accident previously evaluated.

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

The SLMCPR values are designed to ensure that fuel damage from transition boiling does not occur in at least 99.9% of the fuel rods as a result of the limiting postulated accident. The values are calculated in accordance with NRC-approved General Electric methods. The approved General Electric methods are comprehensive for ensuring that fuel designs will perform within acceptable bounds. The SLMCPR ensures that the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. Therefore, the possibility of a new or different kind of accident is not created.

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

The SLMCPR values do not alter the design or function of any plant system. The new values were calculated using NRC-approved methods to maintain the same margin of safety as presently exists for the prevention of transition boiling. At least 99.9% of the fuel rods will avoid transition boiling if the SLMCPR is not violated. Therefore, a significant reduction in a margin of safety is not involved.

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.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: May 10, 1999.

Description of amendment request: The proposed amendment would revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) to: (1)

insert NOTE for Limiting Condition for Operation (LCO) 3.7.4 that would allow intermittent opening of the control building boundary under administrative control; (2) add a CONDITION, REQUIRED ACTION and COMPLETION TIME to LCO 3.7.4 for when both standby filter unit (SFU) subsystems are inoperable due to inoperable control building boundary in MODES 1, 2, and 3; (3) re-letter items in LCO 3.7.4 for consistency; and (4) revise LCO 3.7.4 CONDITION D (new CONDITION E) to add ''for reasons other than CONDITION 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. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Requiring the plant to enter LCO 3.0.3 when the control building pressure envelope is not intact is excessively restrictive. This change provides less restrictive requirements for operation of the facility. These less restrictive requirements do not result in operation that will increase the probability of initiating an analyzed event. The proposed change is acceptable because of the low probability (less than 3.04×10^{-8}) of a DBA [design basis accident] occurring during the 24 hour Completion Time, and the availability the SFU system to provide a filtered environment (albeit with potential control room in-leakage).

Intermittent opening of the control building boundary requires controls which consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated. For entry and exit through doors the administrative control is performed by the person entering or exiting the area. As a result, the consequences of any accident previously evaluated are not significantly increased.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not involve new or different equipment being installed at the facility. The proposed change is acceptable because of the low probability (less than 3.04×10^{-8}) of a DBA occurring during the 24 hour Completion Time, and the availability of the SFU system to provide a filtered environment (albeit with potential control room in-leakage).

Intermittent opening of the control building boundary requires controls which consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control building isolation is indicated. For entry and exit through doors the administrative control is performed by the person entering or exiting the area.

3. The proposed amendment will not involve a significant reduction in a margin of safety. Requiring the plant to enter LCO 3.0.3 when the control room ventilation envelope is not intact is excessively restrictive. The proposed change is acceptable because of the low probability (less than $3.04\times I0^{-8}$) of a DBA occurring during the 24 hour Completion Time.

Intermittent opening of the control room boundary requires controls which consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control building isolation is indicated. For entry and exit through doors the administrative control is performed by the person entering or exiting the area.

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.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: June 8, 1999.

Description of amendment request: The proposed change corrects the described method by which the Standby Gas Treatment system heaters are to be tested. This change is necessary because the reference provided in Technical Specification Section 5.5.7e is in error.

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The correction of an error and clarification of a testing method does not alter any of the precursors assumed in the CNS [Cooper Nuclear Station] accident analysis. The proposed wording for testing SGT [Standby Gas Treatment] heaters is in accordance with ASME N510–1989, Section 14.5.1, ''Testing of Nuclear Air Treatment Systems.'' Since the proposed change does not affect this portion of plant design and operation, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident than evaluated in the Updated Safety Analysis Report (USAR). The proposed change does not result in any physical change to CNS structures, systems, or components, nor does it change the fit, form, or function of any equipment or components taken credit for in the accident analyses described in the USAR. Therefore, correction of a test reference and specific description of the testing method for the SGT heaters does not create the possibility of a new or different kind of accident.

The proposed change will not create a significant reduction in the margin of safety. The proposed change does not alter the design or administrative controls necessary to ensure the required performance of the physical barriers during acticipated operational occurrences and postulated accidents. This conclusion is based on the fact that the proposed change corrects an erroneous reference, conforms to industry standards, and is consistent with past and current operating practice at CNS; therefore, the proposed change does not create 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: 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 Section Chief: Robert A. Gramm.

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

Date of amendment request: June 15, 1999.

Description of amendment request: The proposed change would allow the use of the service water (SW) system to directly supply cooling water to the reactor equipment cooling (REC) system during a loss-of-coolant accident (LOCA) event. The present maximum allowable REC water leakage rate is based on the requirement that there will be sufficient water in the REC surge tank to allow the REC system to fulfill its safety function for 30 days post-LOCA condition. A proposed Updated Safety Analysis Report (USAR) revision would allow Cooper Nuclear Station (CNS) to revise the maximum allowable REC system leakage during normal power operation such that the REC system surge tank would assure that the REC would fulfill its safety function for at least the first 7 days following a large break LOCA. The SW system would fulfill the safety functions of the REC system, if required, for the remaining duration of the 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. The licensee states that the proposed request:

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

The proposed change does not involve a significant increase in the probability of an accident previously evaluated in the USAR since there are no hardware changes associated with this USAR change. Procedure changes associated with this USAR change are limited to direction on which division of SW/REC backup to initiate first, and incorporation of new system leakage limits into surveillance procedures.

The proposed change also does not involve a significant increase in the consequences of an accident previously evaluated in the USAR. This conclusion is based on the safety evaluation (Attachment 2 [of the June 15, 1999, application]) which demonstrates that the SW system will fulfill the safety functions of the REC system in a post LOCA condition and thus the proposed change will not affect the performance and reliability of the REC system. The emergency systems cooled by the REC system, the ECCS [emergency core cooling] systems and their room coolers, will therefore also fulfill their safety function when directly supplied by the SW system.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the USAR. The proposed license amendment does not introduce any new equipment or hardware changes. It does, however, allow the SW system to perform a different type of function than it is presently licensed to perform in a post LOCA condition. This SW system post LOCA function has been previously demonstrated to fulfill the functions of the REC in a non LOCA emergency shutdown which are the same as the functions required following a LOCA.

3. Does not create a significant reduction in the margin of safety.

The proposed activity does not involve a significant reduction in the margin to safety. The safety evaluation (Attachment 2) demonstrates that the SW system will perform the required REC post LOCA functions. There is an added required operator action which is to align the SW system to directly supply cooling water to the REC critical loops. As discussed in the safety evaluation [of the June 15, 1999, application], this action can be performed from the main control room utilizing one control switch and there is sufficient control room indication for the operator to be alerted to the need for the use of service water backup. There is also sufficient time for the operator to perform the task. Trending (prior to a postulated LOCA) routinely provides the control room operator with REC system leakage information. In a post LOCA situation, this leakage information would assist the operator in taking timely action to initiate the service water back-up before the need is alarmed in the control room.

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 Section Chief: Robert A. Gramm.

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

Date of amendment request: June 15, 1999.

Description of amendment request: The purpose of the requested license amendment is to revise the Updated Safety Analysis Report (USAR) to incorporate the latest analysis to demonstrate adequate net positive suction head (NPSH) for the low pressure emergency core cooling system (ECCS) pumps following a large break loss-of-coolant accident (LOCA). Specifically, the change would allow (1) reliance on a slightly larger amount of containment overpressure for residual heat removal (RHR) and core spray (CS) pump operation during worst-case long-term LOCA conditions (greater than 1000 seconds) while still maintaining original license margins of 3 and 6 pounds per square inch (psi), respectively, for the difference between minimum available containment pressure and the pressure required for minimum pump NPSH, (2) reliance on a small amount of containment overpressure for CS pump runout during worst-case short-term LOCA conditions (less than 10 minutes) while still maintaining an adequate pressure margin of at least 5 psi, and (3) the use of ANS 5.1 decay heat model in the USAR Section 5.2.6 as currently presented based on analysis justifying the use of this model as described in the amendment request.

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

The proposed change does not involve an increase in the probability of an accident previously evaluated in the USAR. There are no changes being proposed to the maintenance, operation, or design of plant systems or equipment postulated to initiate accidents or transients.

The proposed change does not involve an increase in the consequences of an accident previously evaluated in the USAR. This conclusion is based on the conclusions of the safety evaluation (Attachment 2 [of the June 15, 1999, application]). This safety evaluation demonstrates that the containment overpressure is sufficiently conservative, and that the calculated margins between the available containment overpressure and the overpressure required to assure adequate low pressure ECCS pump NPSH are such that ECCS pump operation, as credited in the CNS [Cooper Nuclear Station] accident analysis, remains unchanged.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the USAR. The proposed license amendment does not introduce any new equipment or hardware changes. The attached safety evaluation demonstrates that the only equipment affected by this License Amendment are the low pressure ECCS pumps and that these will retain their ability to function following a LOCA.

3. Does not create a significant reduction in the margin of safety.

The proposed activity does not involve a significant reduction in the margin of safety. The safety evaluation (Attachment 2) demonstrates that, although there is an increased reliance on containment overpressure to assure adequate low pressure ECCS pump NPSH, there remains sufficient margin to provide confidence that the ECCS pumps will operate as required. Sufficient margin is demonstrated with the added conservatism of a 2-sigma (2 standard deviation) uncertainty in the decay heat model, increased suction strainer debris loading, increased RHR heat exchanger tube plugging margin, and increases in SW [Service Water] and Suppression Pool temperatures. The minimum margin available between available overpressure and required overpressure is at least 5 psi for CS (just prior to 10 minutes) and at least 3 psi for RHR (well after 10 minutes).

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 Section Chief: Robert A. Gramm.

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

Date of amendment request: June 23, 1999.

Description of amendment request: The proposed change to the Technical Specifications would increase the allowed outage time for the Control Room Air Conditioning Subsystem from 30 days to 60 days, on a one-time basis only, for each train.

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 operational requirements for the Control Room Air Conditioning Subsystems (CRACS) are contained in Technical Specification 3.7.6.2 ''Control Room Subsystems Air Conditioning.'' This Limiting Condition for Operation (LCO) requires that two independent Control Room Air Conditioning Subsystems (trains) be operable during all modes of operation. The LCO action statement for operational modes 1, 2, 3 and 4, with one Control Room Air Conditioning Subsystem inoperable, states: ''restore the inoperable system to operable status within 30 days or be in at least Hot Standby [Mode 3] within the next 6 hours and in Cold Shutdown within the following 30 hours.'' The LCO action statement for operational modes 5 and 6 with one Control Room Air Conditioning Subsystem inoperable, states: ''restore the inoperable system to operable within 30 days or initiate and maintain operation of the remaining OPERABLE Control Room Air Conditioning Subsystem or immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.''

The proposed change adds the following note: ''* For cycle 7, the allowable outage time may be extended to 60 days, on a one-time basis, for each train, to implement modifications to the control room air conditioning subsystems. The provisions of specifications 3.0.4 and 4.0.4 are not applicable during the implementation of modifications to the air conditioning subsystems.''

This change is a one-time only change to Technical Specification 3.7.6.2 in order to facilitate the installation of a design change to the CRACS during the present operating cycle. This change will not affect the existing 30 [day] AOT period presently in place in Technical Specification 3.7.6.2 which requires specific actions in the event that the CRACS is determined to be inoperable for any other reason. The design basis accidents are not affected as a result of the proposed one-time change to the Technical Specifications. The CRACS are support subsystems which can only contribute to the initiation of an accident if the whole function is lost. The plant would be required to shutdown before this occurred. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility (other than the CRACS) or the manner in which the plant is operated nor does it adversely affect the response of the plant to a transient or accident. This one-time change is to be utilized only during the present operating cycle (cycle 7) in order to facilitate the implementation of a design change to modify the existing safety-related refrigerant subsystems (one train at a time) and replace them with safety-related chilled water subsystems. This design change is being implemented to improve the overall reliability of the safety-related subsystems.

The consequences of an extended loss of the operating CRACS and the non-safety related chilled water subsystem, during all modes of operation, would result in a slow gradual rise in control room temperature. The temperature of the control room is normally maintained between 70 to 72°F at the discretion of the Unit Shift Supervisor utilizing a non-safety-related train of CRACS. In the event that the control room temperature increased to a temperature greater than 75°F, plant procedures require starting other equipment in the non-safety-related chilled water subsystem or a safety-related train of CRACS to restore control room temperature to its normal operating band. In the unlikely event that the non-safety-related chilled water subsystems and the operable safety-related train of CRACS fail during the proposed 60 day AOT period, Technical Specification 3.7.6.2 would require that actions be commenced to place the plant in a shutdown condition. Additionally, alternative actions to reduce control room temperature could also be initiated as identified in a plant procedure. It has been conservatively determined that safety-related equipment in the control room can be operated continuously up to 90°F in an environment without affecting the capability of the equipment.

The exception to specifications 3.0.4 and 4.0.4 as stated in the proposed one-time change to Technical Specification 3.7.6.2 will not involve an increase in the probability or consequences of an accident. TS 3.0.4 prohibits entry into a mode when the conditions for the LCO are not met and the associated action(s) requires a shutdown if they are not met within a specified time interval. Surveillance Requirement 4.0.4 prohibits entry into a mode unless the associated surveillance requirement(s) has been performed within the stated interval. During the implementation of the modification, when one safety-related train of CRACS is inoperable, it is possible that a plant shutdown could occur due to reasons unrelated to the planned modifications of the CRACS. As stated above, the CRACS are support subsystems which do not contribute to the initiation of any accident previously evaluated. Entry of the plant into an operational mode from a shutdown mode as a result of the proposed modification does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility (other than the CRACS) or the manner in which the plant is operated nor does it adversely affect the response of the plant to a transient or accident. The functions of the CRACS to provide a controlled environment inside of the control room complex to ensure the comfort of the plant operators and to ensure adequate climate conditions for the operability of equipment will not be impaired in any way as a result of a plant mode change. The remaining actions identified in TS 3.7.6.2 are unchanged as a result of the proposed change. The risk significance involved with removing a safety-related train of the CRACS during power operation or during refueling conditions is low based on the short period (60 days per train) and consequences of losing this function. The CRACS is excluded from modeling in the Seabrook Station Probabilistic Risk Assessment (PRA) due to its extremely low risk significance.

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

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

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated since it is a support system and the loss of function will require a plant shutdown. The proposed change adds the following note which pertains to both affected action statements: ''* For cycle 7, the Allowable Outage Time may be extended to 60 days, on a one-time basis, for each train during the implementation of modifications to the control room air conditioning subsystems. The provisions of specifications 3.0.4 and 4.0.4 are not applicable during the implementation of modifications to the air conditioning subsystems.'' As previously identified, this change is a one-time only change to Technical Specification 3.7.6.2 in order to facilitate the installation of a design change to CRACS during the present operating cycle.

The CRACS are support subsystems which do not contribute to the creation of a new or different kind of accident from any previously evaluated nor is it used to mitigate the consequences of a transient or accident. The functions of the CRACS are to provide a controlled environment inside of the control room complex to ensure the comfort of the plant operators and to ensure adequate climate conditions for the operability of equipment. The CRACS consists of two independent safety-related air conditioning trains that provide cooling of recirculated control room air. Due to previous reliability problems with the CRACS, an additional non-safety chilled water subsystem has been installed to provide control room cooling on a continuous basis. Baseload operation of the non-safety related chilled water subsystem to provide control room cooling reduces the operational load on the safety-related refrigerant trains.

Implementation of the modification to the CRACS subsystems during the 60 day AOT duration in no way affects the availability of the non-safety-related chilled water subsystem or the operable safety-related train of the CRACS to meet the control room cooling requirements. The proposed modification removes freon from the control room complex and the quantity of chilled water in the closed loop system is too small to become a flood hazard. The consequences of an extended loss of the operating CRACS train and the non-safety related chilled water subsystem would result in a slow gradual rise in control room temperature. In the event that control room temperature increased to a temperature greater than 75°F, plant procedures require starting either the non-safety-related chilled water subsystem or a safety-related train of CRACS to restore control room temperature. Additionally, in the unlikely event of a loss of the non-safety related chilled water subsystem and the operable safety-related train of the CRACS, Technical Specification 3.7.6.2 would require that actions be taken to place the plant in a shutdown condition.

It has been conservatively determined that safety-related equipment in the control room can be operated continuously in an environment up to 90°F without affecting the capability of the equipment. This proposed change will not affect the existing 30 day AOT period presently in place in Technical Specification 3.7.6.2 which requires specific actions in the event that the CRACS is determined to be inoperable for any other reason.

The exception to specifications 3.0.4 and 4.0.4 as stated in the proposed one-time change to Technical Specification 3.7.6.2 will not involve the creation of an accident of any type. During the implementation of the proposed modification, when one safety-related train of CRACS is inoperable, it is possible that a plant shutdown could occur due to reasons unrelated to the planned modifications of the CRACS. Entry of the plant into an operational mode from a shutdown mode as a result of the proposed modification does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated nor the manner that it responds to a transient or accident. The functions of the CRACS to provide a controlled environment inside of the control room complex to ensure the comfort of the plant operators and to ensure adequate climate conditions for the operability of equipment will not be impaired in any way as a result of a plant mode change. The remaining actions identified in TS 3.7.6.2 are unchanged as a result of the proposed change.

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

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

The proposed one-time change to Technical Specification 3.7.6.2 will not involve a significant reduction in the margin of safety. The functions of the CRACS are to provide a controlled environment inside of the control room complex to ensure the comfort of the plant operators and to ensure adequate climate conditions for the operability of equipment. The CRACS consists of two independent safety-related air conditioning trains that provide cooling of recirculated control room air. Additionally, the Seabrook Station design incorporates the use of a non-safety chilled water subsystem (which is not within the scope of the Technical Specifications) to provide baseload cooling of the control room on a continuous basis.

Implementation of the modification to the CRACS subsystems during the 60 day AOT duration does not result in a significant reduction in the plant margin of safety. As previously identified, the CRACS is a support subsystem and the existing Technical Specifications will require a plant shutdown on a loss of function. The risk significance involved with removing a safety-related train of the CRACS is extremely low based on the short period (60 days per train) and the consequences of losing this function. The potential that the non-safety-related chilled water subsystem and the operable safety-related train of CRACS simultaneously fail during the proposed 60 day AOT period of each safety-related train (120 days total) is considered unlikely. In the event that control room temperature increased to a temperature greater than 75°F, plant procedures require starting either the non-safety-related chilled water subsystem or a safety-related train of CRACS to restore control room temperature. Additionally, in the unlikely event of a loss of the non-safety related subsystem and the operable safety-related train of the CRACS, Technical Specification 3.7.6.2 would require that actions be taken to place the plant in a shutdown condition. Alternative actions to reduce control room temperature could also be initiated as identified in a plant procedure. It has been conservatively determined that safety-related equipment in the control room can be operated continuously in an environment up to 90°F without affecting the capability of the equipment.

The exception to specifications 3.0.4 and 4.0.4 as stated in the proposed one-time change to Technical Specification 3.7.6.2 will not reduce the margin of safety. During the implementation of the proposed modification, when one safety-related train of CRACS is inoperable, it is possible that a plant shutdown could occur due to reasons unrelated to the planned modifications of the CRACS. Entry of the plant into an operational mode from a shutdown mode as a result of the proposed modification does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The functions of the CRACS to provide a controlled environment inside of the control room complex to ensure the comfort of the plant operators and to ensure adequate climate conditions for the operability of equipment will not be impaired in any way as a result of a plant mode change. The remaining actions identified in TS 3.7.6.2 are unchanged as a result of the proposed change.

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, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 17, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) section 4.4.6.2.2.e to replace the reference to American Society of Mechanical Engineers (ASME) Code paragraph IWV–3472(b) which pertains to the frequency of leakage rate testing for 6-inch, nominal pipe size valves and larger with the requirement that the surveillance interval and frequency of surveillance leakage rate testing for these valves be performed pursuant to the requirements of TS 4.0.5, ''Operations and Surveillance Requirements.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided the NRC 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

Eliminating the reference to ASME Code paragraph IWV–3427(b) and performing pressure isolation valve (PIV) testing pursuant to TS 4.0.5 does not change the test conditions for PIV leakage testing and is consistent with the currently analyzed configurations. This change eliminates an unnecessary test requirement and incorporates Westinghouse Owner's Group (WOG) Standard Technical Specifications (STS) frequency requirements that are deemed to substantially reduce the probability of an intersystem loss-of-coolant-accident. This change in testing frequency requirements does not affect the accident mitigation capabilities of the reactor coolant system (RCS) PIVs. This change is bounded by existing accident analyses. Therefore, it is concluded that, with the reduced probability of previously analyzed accidents, and no effect on accident mitigation, the proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Eliminating the IWV–3427(b) trending for 6-inch and larger valves (and the accompanying increased frequency testing requirement) does not significantly change actual testing frequencies since the frequencies continue to be addressed by the remaining TS requirements. This change does not affect the ability of a PIV to perform its required RCS pressure isolation safety function of limiting RCS leakage to prevent overpressure failure of attached low pressure systems. The frequency of testing or the testing itself are not initiating events to postulated accidents. Therefore, the proposed revision does not create the possibility of a new or different kind of accident from any previously evaluated.

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

There is no impact on the Margin of Safety as defined in the bases of any TS, the Updated Final Safety Analysis Report, or other licensing basis commitments resulting from the elimination of the reference to ASME Code paragraph IWV–3427(b). Periodic surveillances provide continued assurance in the capability of safety related equipment to perform its design safety (accident mitigating) function and are not used to establish the margin of safety for accident mitigation. Therefore, the frequency of surveillance testing of the PIVs has no impact on the margins of safety assumed in analyzed accidents.

In its evaluation, NNECO concluded, based on its evaluation as required by 10 CFR 50.92, that the proposed revision does not involve a SHC.

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 Section Chief: James W. Clifford.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment requests: May 13, 1999.

Description of amendment requests: The proposed amendments would modify Technical Specification (TS) 6.2.A.2, ''Onsite and Offsite Organizations,'' to reflect a change in the organizational structure implemented on March 1, 1999.

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

The proposed change is administrative in nature and does not significantly affect any system that is a contributor to initiating events for previously evaluated accidents. Neither does the change significantly affect any system that is used to mitigate any previously evaluated accidents. Therefore, the proposed change does not involve any significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed change is administrative in nature and does not alter the design, function, or operation of any plant component nor does the proposed change install any new or different equipment, therefore the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

The proposed change is administrative in nature and does not involve a significant reduction in the margin of safety associated with the safety limits inherent in either the fuel cladding, RCS [reactor coolant system] boundary, reactor containment, or other structures, systems, or components (SSCs).

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: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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

NRC Project Director: Claudia M. Craig.

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

Date of amendment request: June 7, 1999.

Description of amendment request: The proposed amendments, if approved, would revise Technical Specifications (TS) Section 3/4.4.3 and its associated TS Bases to reflect changes to refine and clarify the action statement concerning inoperable reactor coolant leakage detection systems.

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

The proposed TS changes directly establish the minimum acceptable level of Reactor Coolant System (RCS) leakage detection instrumentation required to support plant power operations. The level of RCS leakage detection capability inherent with the proposed TS change will continue to provide acceptable early warning detection of potential RCS pressure boundary degradation as required under 10 CFR 50.36 (c)(2)(ii) (A) Criterion 1.

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

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

The proposed TS changes only affect systems associated with the detection of accidents involving degradation of the RCS pressure boundary. The proposed TS changes do not involve any physical changes to plant structures, systems, or components. The RCS Leakage Detection Systems will continue to function as designed in all modes of operation. No new accident type is created as a result of the proposed changes. No new failure mode for any equipment is created. The changes are consistent with the guidance provided in [Standard Technical Specifications General Electric Plants BWR/4 dated April 1995] NUREG–1433, Revision 1, pertaining to RCS Leakage Detection.

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

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

The TS Limiting Conditions for Operation (LCO) specify for systems and equipment important to safety, the minimum level of operability required to permit continued power operation. The proposed TS changes revise this minimum level of operability by permitting long term plant operation with the removal of the Drywell Unit Coolers Condensate Flow Rate Monitoring System from service. Currently, this condition would permit the plant to continue to operate for up to 30 days. This change is not a reduction in the margin of safety since:

The proposed Technical Specification LCO change for RCS Leakage Detection Systems maintains four (4) diverse methods of detecting RCS leakage and permits continuous operation with the Drywell Unit Coolers Condensate Flow Rate Monitors out of service provided that more frequent surveillance checks are provided for the containment atmosphere monitoring system. The proposed TS change institutes the additional surveillance requirements.

The LGS reactor coolant pressure boundary was designed to ASME Class 1, Seismic Category I design criteria with no special dispensation which would warrant such additional RCS leakage detection capability or more stringent LCO criteria than those generically approved under the Improved Standard Technical Specifications.

Review of the TS Bases Section and UFSAR identified no discussions regarding margin of safety for the RCS Leakage Detection Systems, which would be reduced by the proposed Technical Specification LCO change. It is further demonstrated that an acceptable margin of safety exists based on the generic regulatory approval of the Improved Standard Technical Specifications which will remain bounded by the proposed LGS TS changes.

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

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

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

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

NRC Section Chief: James W. Clifford.

Portland General Electric Company, Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: August 27, 1998.

Description of amendment request: The proposed amendment would revise the Facility Operating (Possession-Only) License and the Permanently Defueled Technical Specifications. Multiple license conditions and technical specification requirements are proposed to be deleted to reflect the transfer of the nuclear spent fuel from the existing 10 CFR Part 50 licensed area to the 10 CFR Part 72 Independent Spent Fuel Storage Installation (ISFSI) area.

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.

This proposed amendment reflects removal of the spent nuclear fuel from the 10 CFR 50 licensed area and transfer of the spent nuclear fuel to the 10 CFR 72 ISFSI licensed area. The probability and consequences of accidents associated with storage of spent nuclear fuel within the TNP [Trojan Nuclear Plant] ISFSI were evaluated as part of PGE's 10 CFR 72 license application. Following completion of the transfer of the spent nuclear fuel to the 10 CFR 72 licensed ISFSI and in light of the revised Appendix A Technical Specification, Section 4.2, that precludes storage of spent nuclear fuel within the 10 CFR 50 licensed area, the potential for accidents associated with the storage and handling of fuel in the 10 CFR 50 licensed area will be eliminated. Therefore, deleting those technical specifications associated with spent nuclear fuel will not result in a significant increase in the probability or consequences of accidents previously analyzed.

The proposed license amendment also relocates administrative requirements from Section 5.0 of the Technical Specifications to topical report PGE–8010, ''TNP Nuclear Quality Assurance Program.'' Relocation of administrative requirements follows the guidance provided in NRC Administrative Letter 95–06. Relocating these administrative requirements will not result in changes in method of operation of any plant equipment, therefore these changes will not result in a significant increase in the probability or consequences of accidents previously evaluated.

The proposed license amendment will delete the on duty shift manning requirements (Technical Specification 5.2.2a). With removal of the spent nuclear fuel from the 10 CFR 50 licensed area, there are no remaining important to safety systems required to be monitored. With removal of the spent nuclear fuel from the 10 CFR 50 licensed area, there are no remaining credible accidents which require the actions of a Shift Manager or non-certified operator to prevent occurrence or mitigate consequences. Therefore, deleting the shift manning requirements will not result in an increase in the probability or consequences of an accident previously analyzed.

Deleting the Independent Review and Audit Committee (IRAC) is also proposed in this license amendment request. The responsibility of IRAC is to review and advise the Plant General Manager on matters relating to the safe storage of irradiated fuel. Since approval of this license amendment request is contingent upon removal of the spent nuclear fuel from the 10 CFR 50 licensed area and a revised Technical Specification Section 4.2 prevents future storage, deleting IRAC will not result in an increase in the probability or consequences of an accident previously evaluated.

This license amendment request proposes to revise and relocate License Condition 2.C.(8), Fire Protection, to the TNP Quality Assurance Program (PGE-8010). The revised text removes requirements associated with making changes that could adversely impact the safe storage of irradiated fuel. Following removal of the spent nuclear fuel from the 10 CFR 50 licensed area and implementation of the proposed revision to Technical Specification Section 4.2, irradiated fuel will not be stored within the 10 CFR 50 licensed area so this change will not result in an increase in the probability of occurrence or consequences of accidents previously analyzed. Relocation of the remaining requirements contained in this license condition to the TNP Quality Assurance Program (PGE–8010) will provide the necessary administrative control to ensure that changes to the fire protection program will not increase the likelihood of an offsite release of radioactive material due to a fire. Therefore, this change will not result in an increase in the probability of occurrence or consequence of accidents previously analyzed.

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

The proposed license amendment reflects the reduced operational risks within the 10 CFR 50 licensed area after the spent nuclear fuel has been transferred to the ISFSI. In addition, administrative controls contained in Section 5.0 of the Technical Specifications will [be] relocated to PGE–8010, TNP Nuclear Quality Assurance Program. These changes have no impact on plant equipment and only an administrative impact on some of the procedures used for operating plant equipment, which may still be needed within the 10 CFR 50 licensed area following the transfer of the spent nuclear fuel to the 10 CFR 72 ISFSI license area. This proposed amendment does not result in the addition of new equipment or result in the alteration of the operation of existing structures, systems, or components. 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 10 CFR 50 license amendment eliminates those technical specifications and license conditions associated with the storage of spent nuclear fuel. Following transfer of the spent nuclear fuel to the 10 CFR 72 ISFSI, the potential for fuel related accidents will be eliminated from the 10 CFR 50 licensed area. Therefore, removal of those technical specifications and license conditions associated with the safe storage of spent nuclear fuel will not involve a significant reduction in a margin of safety.

Relocating administrative programs in Technical Specification, Section 5.0, ''Administrative Controls,'' follows the guidance of NRC Administrative Letter 95–06. With the exception of deleting those administrative controls associated with storage of spent nuclear fuel, the administrative programs will be relocated to the TNP Quality Assurance Program (PGE–8010). This administrative relocation of requirements does not involve a significant reduction in a margin of safety.

This proposed amendment also requests deleting several license conditions and relocating License Condition 2.C.(8), ''Fire Protection.'' The deleted license conditions were related to either power operations or activities which have been completed and are no longer required. Relocating License Condition 2.C.(8), ''Fire Protection,'' to the TNP Quality Assurance Program (PGE–8010) will continue to maintain the required level of administrative control for the fire protection program since changes to PGE–8010 are controlled in accordance with the requirements of 10 CFR 50.54(a)(3). Deleting these license conditions will, therefore, not result in a reduction in the margin of safety.

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

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Attorney for licensee: Leonard A. Girard, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Michael T. Masnik.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: June 7, 1999, as supplemented by letter dated June 24, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 2.0, Safety Limits and Limiting Safety System Settings, TS 3.2.5, DNB [Departure from Nucleate Boiling] Parameters, and the associated Bases, and Administrative Controls Section 6.9.1.6, Core Operating Limits Report (COLR), by relocating cycle-specific reactor coolant system-related parameter limits from the TSs to the COLR. This would allow for flexibility to enhance plant operating margin and/or core design margins without the need for cycle-specific license amendment requests.

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

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

The proposed amendment is a programmatic and administrative change that does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve an increase in the probability or consequences of any accident previously evaluated.

The cycle-specific limits in the Core Operating Limits Report will continue to be controlled by the STP [South Texas Project] programs and procedures. Each accident analysis addressed in the UFSAR [Updated Final Safety Analysis Report] will be examined with respect to changes in the cycle-dependent parameters, which are obtained from the use of NRC-approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be conducted per the requirements of 10CFR50.59, will ensure that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The safety limits imposed in Technical Specification 2.1.1.1 and 2.1.1.2 are consistent with the values stated in the STP Updated Final Safety Analysis Report. The Reactor Coolant System Flow value in the Technical Specifications will be changed from the Minimum Measured Flow to the Thermal Design System Flow (approved by the Nuclear Regulatory Commission in Amendments 97 and 84 on September 29, 1998) consistent with WCAP–14483–P–A ['Generic Methodology for Expanding Core Operating Limits Reports']. This change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

Removal of cycle specific variables has no influence or impact on, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The cycle specific variables are calculated using the NRC-approved methods, and submitted to the NRC to allow the staff to continue to trend the values of these limits. The Technical Specifications will continue to require operation within the core operating limits, and appropriate actions will be required if these limits are exceeded. The safety limits imposed in Technical Specification 2.1.1.1 and 2.1.1.2 are consistent with the values stated in the STP Updated Final Safety Analysis Report. The Reactor Coolant System Flow value in the Technical Specifications will be changed from the Minimum Measured Flow to the Thermal Design Flow (approved by the Nuclear Regulatory Commission in Amendments 97 and 84 on September 29, 1998) consistent with WCAP–14483–P–A. This proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is not affected by the removal of cycle specific core operating limits from the Technical Specifications. The margin of safety presently provided by current Technical Specifications remains unchanged. Appropriate measures exist to control the values of these cycle specific limits. The proposed amendment continues to require operation within the core limits as obtained from NRC-approved reload design methodologies, and the actions to be taken if a limit is exceeded remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC-approved documentation. In addition, each future reload will involve a 10CFR50.59 safety review to assure that operation of the unit within cycle-specific limits will not involve a significant reduction in the margin of safety.

The safety limits imposed in Technical Specification 2.1.1.1 and 2.1.1.2 are consistent with the values stated in the STP Updated Final Safety Analysis Report. The Reactor Coolant System Flow value in the Technical Specifications will be changed from the Minimum Measured Flow to the Thermal Design System Flow (approved by the Nuclear Regulatory Commission in Amendments 97 and 84 on September 29, 1998) consistent with WCAP–14483–P–A. This proposed change does not involve a significant reduction in the margin of safety.

The proposed amendment is a programmatic and administrative change that provides assurance that plant operations continue to be conducted in a safe manner. As stated previously, the proposed amendment does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. Because the design of the facility and system operating parameters are not being changed, the proposed amendment 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 3, 1999.

Description of amendment request: The proposed amendment would modify the Technical Specifications to reduce the Allowable Value (Av) used for Reactor Vessel Water Level—Low, Level 3 for several instrument functions.

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

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

The Reactor Vessel Water Level—Low, Level 3 functions are in response to water level transients and are not involved in the initiation of accidents or transients. Therefore, reducing the Level 3 Av does not increase the probability of an accident previously evaluated. Additionally, the results of the safety evaluation associated with the lowering of the Level 3 Av concludes that the previously evaluated transient and accident consequences are not significantly affected by the change. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment to lower the BFN Units 2 and 3 Reactor Vessel Water Level—Low, Level 3 Av does not involve a hardware change and the purpose of the Level 3 function is not affected. The Level 3 functions will continue to fulfill their design objective. Therefore, reduction of the Av does not result in the possibility of a new or different kind of accident.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The results of the safety evaluation associated with the reducing the BFN Units 2 and 3 Reactor Vessel Water Level—Low, Level 3 Av concluded that transient and accident consequences remain within the required acceptance criteria. Therefore, the margin of safety is not reduced for any event evaluated.

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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: June 23, 1999.

Brief description of amendments: The proposed license amendments would change the way in which the Emergency Diesel Generator (EDG) automatic trips are tested in Surveillance Requirement (SR) 3.8.1.13. A note would also be added to specify the CPSES, Unit 2, test schedule in SR 3.8.1.13.

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

The emergency diesel generators are used to support mitigation of the consequences of an accident and are not considered to be initiator of any previously analyzed accident. Revising the surveillance to verify the bypass of non-critical EDG trips on both LOOP [loss of offsite power] and SI [safety injection] separately enhances the ability of the EDG to perform its safety function by ensuring continued operation during DBAs [design-basis accidents].

Therefore, this change will not result in an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the surveillance requirement involves an EDG start circuit modification. The circuit modification has been previously installed on Unit 2 during 2RF04 [CPSES Unit 2, fourth refueling outage] for reasons other than the issue associated with the FWLB [feedwater line break]. As a part of the Unit 2 installation a 50.59 evaluation was performed and it was determined that the modification did not represent an unreviewed safety question. The modification similar to Unit 2 will be implemented on Unit 1 and therefore, as concluded in the safety evaluation for the original modification, no new failure mechanisms will be introduced by the proposed change. The EDGs are designed to provide electrical power to equipment important to safety in the event of a loss of offsite power. The proposed change to the SR enhances the confidence that the EDGs will start and fulfill their safety related function.

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

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

The proposed change will not alter any accident analysis assumptions, initial conditions, or results. Revising the surveillance requirement to verify the EDG trip bypass for the LOOP and SI separately will enhance the confidence that the EDG starts as assumed in the safety analyses and does not create any new failure scenarios and no margin is reduced.

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: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gram.

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

Date of amendment request: April 16, 1999, as superseded on June 9, 1999.

Description of amendment request: The licensee proposed clarifying the inservice inspection requirements for Vermont Yankee Nuclear Power Station regarding the granting of relief from ASME Code requirements by the NRC. The licensee also proposed changes to reflect the previous NRC approval of the use of ASME Code Case N–560 at Vermont Yankee Nuclear Power Station.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated?

This change is only an administrative change that: (1) clarifies the NRC's authority to grant relief to a specific requirement, and (2) conforms the TS language regarding GL 88–01 to agree with the NRC's acceptance of ASME Code Case N–560 for use at VY. This conclusion is justified in that:

(a) The pursuit of relief from the ASME code and the imposition of alternative requirements are governed by 10CFR50.55a and require NRC approval. There are several sections in the regulations under which such relief can be granted. The removal of reference to a specific section of CFR that may be used to grant relief has no effect on plant equipment or its operation.

(b) Adding words to clarify the relationship between GL 88–01 and Code Case N–560 eliminates a contradiction in sample selection criteria and does not affect any equipment or its operation.

These changes can be considered administrative in nature and do not change any of the accident analyses for the facility. Thus, there are no changes to the probability or consequences of accidents previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The revision of the wording in the TS to generalize the granting of relief to the ASME code does not result in any changes to the plant equipment or its operation. Similarly, adding words to allow use of the NRC-approved alternative to the sample selection guidance provided in GL 88–01 does not impact plant equipment or its operation. These changes are administrative in nature and do not result in the creation of any new or different kinds of accidents.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

This change primarily revises the wording in the TS to clarify the NRC's authority to grant relief to ASME Section XI requirements. The change maintains the requirement for NRC approval to be obtained for such relief. Secondly, this change conforms the TS language regarding GL 98–01 to agree with a previous relevant NRC disposition [Reference (e)]. [The staff notes that reference (e) is an NRC letter dated November 9, 1999, which approved the use of Code Case N–560 at Vermont Yankee Nuclear Power Station.] These administrative changes do not result in a reduction in 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 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 24, 1999

Description of amendment request: The amendment clarifies the basis for the reactor protection system bypass of the turbine stop valve (TSV) closure and turbine control valve (TCV) fast closure scram signals at low power. The amendment clarifies that the analytical basis for this bypass corresponds to a fraction of reactor rated thermal power and not other measures of power, for instance, turbine power.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change clarifies the basis for the reactor protection system bypass of the turbine stop valve closure and turbine control valve fast closure scram signals. Consideration of the bypass function itself only applies to certain pressurization transients and not accident analyses.

The change properly states the basis for the scram bypass and relates it to reactor thermal power and precludes potential misinterpretation of the basis for the bypass setpoint. Turbine power lags reactor power over the range of concern. Therefore, changing terminology related to ''power'' to mean ''reactor power'' instead of ''turbine power'' is conservative. Accordingly, this change can not be less restrictive.

The low power (TSV closure and TCV fast closure) scram signal bypass does not initiate or mitigate any accident considered in the Updated Final Safety Analysis Report. This function is enabled at higher power to mitigate the effects of the pressurization transient which results from TSV closure or TCV fast closure. This change will not alter assumptions relative to the initiation or mitigation of any accident event.

This change will not involve a significant increase in the probability or consequences of an accident previously evaluated since there is no physical alteration of the plant configuration or relaxation of setpoints or operating parameters.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The reactor protection system bypass of the turbine stop valve closure and turbine control valve fast closure scram signals is not considered an initiator of any accident. This change to clarify the basis for applicability of the bypass does not create any new or different kind of accident since it does not involve any change in the physical configuration of the plant, nor relaxation of setpoints or operating parameters.

VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the change merely adds a more restrictive interpretation to current terminology.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The change involves reducing the potential for misinterpreting the basis for the reactor protection system bypass of the turbine stop valve closure and turbine control valve fast closure scram signals and consequent potential for nonconservative operation of the plant. As a result, the potential for operation of the plant in an unsafe condition is reduced, thereby maintaining the margin of safety.

VY has determined that the proposed change does not involve a significant reduction in a margin of safety since operation of the plant consistent with analytical bases of operation is further assured.

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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and

Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

Notice of Issuance of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: March 30, 1999.

Brief description of amendments: The amendments revised license conditions in each of the operating licenses to delete those license conditions that no longer apply, make an editorial change in the Unit 1 license, and provide clarifying information regarding the license condition in each license concerning equalizer valve restrictions.

Date of issuance: June 25, 1999.

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 188 & 185.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the licenses.

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24195).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 25, 1999.

No significant hazards consideration comments received: No.

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

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: March 30, 1999.

Brief description of amendment: The amendment adds Section 4.0.2 to allow a 24-hour grace period for performing inadvertently missed surveillance.

Date of issuance: June 25, 1999.

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

Amendment No.: 202.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 19, 1999 (64 FR 27317).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 25, 1999.

No significant hazards consideration comments received: No.

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

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: March 30, 1999.

Brief description of amendment: The amendment adds Section 4.0.2 to allow a 24-hour grace period for performing inadvertently missed surveillance.

Date of issuance: June 25, 1999.

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

Amendment No.: 202.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 19, 1999 (64 FR 27317).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 25, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 6, 1998, as supplemented by letter dated May 18, 1999.

Brief description of amendment: The amendment approves a change to Technical Specification (TS) 3.1.3.2, ''Position Indicator Channels—Operating,'' which adopts requirements that are consistent with NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' In addition, the amendment approves the relocation of TS Table 3.8–1, ''Containment Penetration Conductor Overcurrent Protective Devices,'' to licensee control procedures in accordance with the guidance provided in Generic Letter 91–08, ''Removal of Component Lists From Technical Specifications.''

Date of issuance: June 29, 1999.

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

Amendment No.: 208.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56245).

The May 18, 1999, letter provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 17, 1996, as supplemented by letters dated October 22, 1998, and January 12 and February 5, 1999.

Brief description of amendment: The amendment extends the surveillance test interval for the reactor trip circuit breakers from monthly to quarterly and revises the appropriate Bases page.
Date of issuance: June 29, 1999.
Effective date: As of the date of issuance and shall be implemented 60 days from the date of issuance.
Amendment No.: 153.
Facility Operating License No. NPF–38: Amendment revised the Technical Specifications.
Date of initial notice in **Federal Register**: September 9, 1998 (63 FR 48261).
The October 22, 1998, and January 12 and February 5, 1999, letters provided additional information that did not extend the scope of the original no significant hazards consideration determination.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1999.
No significant hazards consideration comments received: No.
Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: October 27, 1998, supplemented March 19, 1999.
Brief description of amendment: This amendment relocates a TS surveillance requirement from TS Section /4.6.5.1, ''Shield Building—Emergency Ventilation System'' to TS Section 3/4.6.5.2, ''Shield Building Integrity.'' Administrative and bases changes have also been made.
Date of issuance: June 22, 1999.
Effective date: June 22, 1999.
Amendment No.: 233.
Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.
Date of initial notice in **Federal Register**: November 18, 1998 (63 FR 64125). The March 19, 1999, supplement to the application did not expand the scope of the original application as noticed, and did not change the staff's proposed no significant hazards consideration determination.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 22, 1999.
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.

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

Date of application for amendment: August 31, 1998, as revised on March 18, 1999.
Brief description of amendment: The amendment approves changes to the Improved Technical Specifications to allow a repair roll process which would be used to repair steam generator tubes with defects within the upper tubesheet. Changes to inservice inspection and reporting requirements and several format and editorial changes were also included.
Date of issuance: June 28, 1999.
Effective date: As of date of issuance, to be implemented prior to commencing Cycle 12 operation.
Amendment No.: 179.
Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.
Date of initial notice in **Federal Register**: October 21, 1998 (63 FR 56249). The revised submittal dated March 18, 1999, expanded the scope of the amendment request as originally noticed, and the application was renoticed on April 21, 1999 (64 FR 19557).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 28, 1999.
No significant hazards consideration comments received: No.
Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: February 7, 1997, as supplemented October 24, 1998
Brief description of amendment: The amendment incorporates changes to more accurately reflect current plant design, adopts changes in surveillance requirements consistent with the Standard Technical Specifications, identifies changes to plant systems and revisions to Technical Specifications system descriptions not involving Limiting Conditions for Operations, and makes editorial or typographical corrections.
Date of issuance: June 21, 1999.
Effective date: As of the date of issuance, and shall be implemented within 30 days.
Amendment No.: 212.
Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.
Date of initial notice in **Federal Register:** March 25, 1998 (63 FR 14486) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 21, 1999.
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.

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

Date of application for amendment: June 11, 1998.
Brief description of amendment: The amendment revises Technical Specification 6.12.1 to allow use of an alternative high radiation area control consistent with Regulatory Guide 8.38.
Date of issuance: July 1, 1999.
Effective date: As of the date of issuance and shall be implemented within 30 days.
Amendment No.: 213.
Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.
Date of initial notice in **Federal Register:** August 12, 1998 (63 FR 43204) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 1, 1999.
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.

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

Date of application for amendment: December 22, 1998, as supplemented March 19, 1999.
Brief description of amendment: The amendment replaces specific titles in Section 6.0 of the Technical Specifications of all three Millstone units with generic titles.
Date of issuance: June 3, 1999.
Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance.
Amendment No.: 105, 235, and 171.

Facility Operating License Nos. DPR–21, DPR–65, and NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 27, 1999 (64 FR 4158). The March 19, 1999, letter provided clarifying information that did not change the scope of the December 22, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 3, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 4, 1999, as supplemented April 7, 1999.

Brief description of amendment: The amendment changes Technical Specifications 3.5.2, ''Emergency Core Cooling Systems—ECCS Subsystems—Tavg greater than or less than 300 °F;'' 3.6.2.1, ''Containment Systems—Depressurization and Cooling Systems—Containment Spray and Cooling Systems;'' 3.7.1.2, ''Plant Systems—Auxiliary Feedwater Pumps;'' 3.7.3.1, ''Plant Systems—Reactor Building Closed Cooling Water System;'' and 3.7.4.1, ''Plant Systems—Service Water System.'' The changes were made to the system pump flow requirements to incorporate the results of revised hydraulic and accident analyses.

Date of issuance: June 29, 1999.

Effective date: As of the date of issuance and shall implemented within 60 days from the date of issuance.

Amendment No.: 236.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 14, 1999 (64 FR 2523).

The April 7, 1999, supplemental letter did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 1999.

No significant hazards consideration comments received: No.

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

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: March 9, 1999, as supplemented May 3, 1999

Brief description of amendments: The amendments delete the requirement to have an independent safety engineering group (ISEG) from the Technical Specifications and applies the substantive requirements now applicable to the ISEG to other organizations and relocates those requirements from the Technical Specifications to Chapter 16 of the Operational Quality Assurance Plan (OQAP). In the letter of May 3, 1999, the licensee submitted the changes to Chapter 16 of the OQAP to incorporate the substantive Technical Specification requirements currently applicable to the ISEG into the OQAP in the form of an independent technical review program, and stated that these changes to the OQAP will become effective upon approval of the amendments.

Date of issuance: June 23, 1999. Effective date: June 23, 1999, to be implemented within 30 days. Implementation includes incorporating the OQAP pages into the OQAP.

Amendment Nos.: Unit 1–112 ; Unit 2–99.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 7, 1999 (64 FR 17030) The May 3, 1999, supplement provided additional clarifying information within the scope of the original notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 23, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 27, 1999, as supplemented May 28, 1999.

Brief description of amendments: The amendments add a footnote to Technical Specfiication 4.8.2.1e, ''D.C. Sources—Operating,'' which would, on a one-time basis for Unit 1 Battery BT1ED2, allow TU Electric to substitute a performance discharge test ''...in lieu of the battery service test required by Specification 4.8.2.1d, twice within a 60 month interval.''

Date of issuance: June 28, 1999.

Effective date: As of the date of issuance.

Amendment Nos.: 65 and 65

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (64 FR 31881 dated June 14, 1999). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by July 14, 1999, but indicated that if the Commission makes a final determination, any such hearing would take place after issuance of the amendments.

The May 28, 1999, letter provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and final NSHC determination are contained in Safety Evaluation dated June 28, 1999.

Attorney for Licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC, 20036.

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

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: February 16, 1999.

Brief Description of amendments: These amendments revise TS Section 4.2 for Units 1 and 2. The changes relax the surveillance requirements for reactor coolant pump (RCP) flywheels. The flywheels provide extended reactor coolant flow coastdown capability if electric power for the RCPs is lost. Previously, the flywheel inspections included an ultrasonic examination (UT) of areas of high stress concentration at the base and keyway every 3 years, and complete UT every 10 years. The changes require only a 10-year UT based upon an analysis presented in a Westinghouse topical report which has been reviewed and accepted by the NRC staff.

Date of issuance: July 1, 1999.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: 221 and 221.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register**: May 5, 1999 (64 FR 24204). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 1, 1999.

No significant hazards consideration comments received: No.

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

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By August 13, 1999, 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Unit 2, Matagorda County, Texas

Date of amendment request: July 1, 1999.

Brief description of amendment: The amendment provides for a one-time change to Technical Specifications 3.3.2 and 3.7.8 for Unit 2 to allow all fuel handling building exhaust air system components to be inoperable for a period not to exceed 8 hours to facilitate repair of the Train B exhaust booster fan.

Date of issuance: July 2, 1999.

Effective date: From the date of amendment issuance until July 14, 1999.

Amendment No.: Unit 2–100.

Facility Operating License No. NPF–80: The amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated July 2, 1999.

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

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

NRC Section Chief: Robert A. Gramm.

Dated at Rockville, Maryland, this 7th day of July 1999.

For The Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–17750 Filed 7–13–99; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified by its task number, DG–3014 (which should be mentioned in all correspondence concerning this draft guide), is a proposed Revision 1 of Regulatory Guide 3.66, ''Standard Format and Content of Financial Assurance Mechanisms Required for Decommissioning Under 10 CFR Parts 30, 40, 70, and 72.'' This proposed revision is being developed to update the NRC's guidance on how to demonstrate financial assurance for decommissioning. The guide also establishes a standard format for presenting the information to the NRC.

The draft guide has not received complete staff approval and does not represent an official NRC staff position.

Comments may be accompanied by relevant information or supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW, Washington, DC. Comments will be most helpful if received by September 30, 1999.

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415–5905; e-mail *CAG@nrc.gov*. For information about the draft guide and the related documents, contact Mr. L.M. Bykoski, (301) 415–6754; e-mail LMB1@nrc.gov.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or