byproduct material generated at its ISL facility at an NRC-approved byproduct disposal facility.

Offsite environmental impacts are related to: (1) Effects on the regional groundwater system, and (2) the potential for increased radiological doses to the general public. Because the issues associated with impacts on the regional groundwater system concern consumptive water use, the NRC has referred further assessment of these impacts to the State of Nebraska. The NRC anticipates that these issues would be addressed by the State at such time as Crow Butte applies for a modification to its Underground Injection Control permit with the State, for a corresponding increase in processing flow rate.

Although the estimated radon release associated with a processing flow rate of 5000 gpm is slightly higher than previously approved, the NRC staff concluded that the modeling satisfactorily shows that the potential impacts to offsite individuals remain well below the 100 mrem/yr (1 mSv/yr) public dose limit of 10 CFR 20.1301. The largest dose estimate was 20.3 mrem/yr (0.203 mSv/yr) for the receptor located approximately 1.0 kilometer from the processing plant vent location.

Conclusion

The NRC staff concludes that approval of Crow Butte's amendment request to increase the processing flow rate at its ISL facility from 3500 gpm to 5000 gpm will not cause significant environmental impacts.

Alternatives to the Proposed Action

Since the NRC staff has concluded that there are no significant environmental impacts associated with the proposed action, any alternatives with equal or greater environmental impacts need not be evaluated. The principal alternative to the proposed action would be to deny the requested action. Since the environmental impacts of the proposed action and this no-action alternative are similar, there is no need to further evaluate alternatives to the proposed action.

Agencies and Persons Consulted

The NRC staff consulted with the State of Nebraska, Department of Environmental Quality (NDEQ), in the development of the Environmental Assessment. A facsimile copy of the final Environmental Assessment was transmitted to Mr. Frank Mills of the NDEQ on January 3, 1996. In a telephone conversation on January 11, 1996, Mr. Mills indicated that the NDEQ had no comments on the Environmental Assessment.

Finding of No Significant Impact

The NRC staff has prepared an Environmental Assessment for the proposed amendment of NRC Source Material License SUA–1534. On the basis of this assessment, the NRC staff has concluded that the environmental impacts that may result from the proposed action would not be significant, and therefore, preparation of an Environmental Impact Statement is not warranted.

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

Notice of Opportunity for Hearing

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

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

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

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

(1) The applicant, Crow Butte Resources Inc., 216 Sixteenth Street Mall, Suite 810, Denver, CO 80202;

(2) The NRC staff, by delivery to the Executive Director of Operations, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, or by mail addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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

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

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

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

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

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

Dated at Rockville, Maryland, this 21st day of February 1996.

For the Nuclear Regulatory Commission.

Daniel M. Gillen,

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

[FR Doc. 96–4483 Filed 2–27–96; 8:45 am]

BILLING CODE 7590–01–P

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 5, 1996, through February 15, 1996. The last biweekly notice was published on February 14, 1996 (61 FR 5809).

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

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

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

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

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

By March 29, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public

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

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

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

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

Date of amendments request: December 20, 1995.

Description of amendments request: The proposed amendment would change the instrumentation setpoint for the reactor trip and main steam isolation signal (MSIS) actuation on low steam generator pressure from greater than or equal to 919 psia with an allowable value of greater than or equal to 911 psia to greater than or equal to 895 psia with an allowable value of greater than or equal to 890 psia.

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

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

The proposed amendment does not involve any change to the method of operation of any plant equipment that is used to mitigate the consequences of an accident. The proposed change only affects the instrument setpoint for steam generator low pressure reactor trip and MSIS actuation. The proposed setpoint meets the requirement of ensuring a reactor trip and MSIS actuation prior to steam generator pressure reaching the analytical limits even under worst-case accident conditions. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed amendment does not alter any of the assumptions or bounding conditions currently in the UFSAR [updated final safety analysis report] and meets the requirement of ensuring a reactor trip and MSIS actuation prior to steam generator pressure reaching the analytical setpoint under worst-case accident conditions. As a result, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve any change to the method of operation of any plant equipment that is used to mitigate the consequences of an accident. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed change. The intent of the proposed change is to increase the margin between normal operating parameters and trip setpoints. This minimizes the possibility of unnecessary challenges to safety systems improving the safety of operation. The method of protecting the facility for an excess steam demand event remains unchanged and therefore, the amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is the implementation of a setpoint value which was derived using methodologies endorsed by Revision 2 of NRC Regulatory Guide 1.105, ''Instrument Setpoints.'' The new setpoint ensures that sufficient margin exists below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection (reactor trip and main steam line isolation) in the event of an excessive steam demand event. The new setpoint ensures that safety margins are maintained within the results of existing calculations. The margin of safety between the analyzed trip value and the point at which safety analysis results become unacceptable remain unchanged since the analytical setpoints are not affected by the amendment. The new setpoint resulted from the reduced instrument uncertainty and will ensure that the reactor trip and MSIS actuation on low steam generator pressure will occur before the analyzed value and hence, this change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

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

NRC Project Director: William H. Bateman.

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

Date of amendments request: January 5, 1996.

Description of amendments request: The proposed amendment would revise paragraph 2.C.(1) of the operating licenses and Section 1.26 of the TS for each of the three PVNGS Units to increase the authorized 100 percent reactor core power (rated thermal power) from 3800 megawatts thermal (Mwt) to 3876 Mwt, an increase of 2 percent. The proposed amendment would also revise TS 4.1.1.4, TS 3.1.3.4, and TS 3.2.6 (Figure 3.2–1) to lower the allowable reactor coolant system cold leg temperature limits for each of the three PVNGS Units, and revise TS 3.4.2.1 and TS 3.4.2.2 to lower the pressurizer safety valve setpoints for Units 1 and 3 to support the increased power operation.

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

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

The proposed amendment does not change the method of operation or modify the plant configuration other than minor changes in equipment setpoints. Thus no increase in the probability of an accident is created by this amendment. System and programmatic reviews have been performed on the nuclear steam supply system controls, reactor coolant system mechanical, steam generator mechanical, balance of plant systems, and fire protection, equipment qualification, and probabilistic risk assessment programs. The conclusion of these reviews was that operation in accordance with the changes proposed in this amendment was acceptable and posed no significant risk to the health and safety of the public. The analyses supporting this amendment demonstrate that the consequences of events using the changes specified in the amendment are within the criteria which are the current licensing basis for the PVNGS Units. Therefore the amendment, as proposed, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment does not modify the configuration of the units except for minor equipment setpoints. No equipment changes and no new methods of plant operation are being proposed, therefore, no new failure modes are introduced by the proposed amendment. The setpoint changes proposed have been evaluated and shown to be acceptable in providing their design function. The increased rated thermal power and associated changes have been incorporated into the safety analysis performed in support of this amendment request and the results have been shown to be similar to those previously obtained. No possibility of a new or different kind of accident from any accident previously evaluated will be created as a result of the proposed amendment.

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

The changes proposed were evaluated in the safety analysis performed to justify the amendment request. Although the consequences of some events increased slightly, the results continue to meet the criteria which form the PVNGS licensing basis. The programmatic and system reviews provide further assurance of the capability of the units to continue to operate safely with the changes proposed in this amendment. Therefore the amendment does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

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

NRC Project Director: William H. Bateman.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: January 29, 1996.

Description of amendment request: The proposed change would revise the technical specifications (TS) table 4.1–3, item 4 to change the frequency of main steam safety valve (MSSV) testing to that specified in NUREG–1431, the improved ''Standard Technical Specifications, Westinghouse Plants'' (one third of the MSSVs each refueling outage). In addition, the licensee proposed adding the MSSV test acceptance 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:

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

Neither the valves' nor the system's configuration or functions are being altered. The valves' setpoints and their ''as-left'' range, +/−1%, will not be changed. The changes are to the testing frequency and the ''as found'' tolerance of the MSSV setpoint.

The proposed changes in testing frequency and the higher tolerance are in the less conservative direction, but are not significant for several reasons. First, the new standards are based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The new standards have been accepted by the nuclear industry and the NRC, and are referenced in the improved Standard Technical Specifications. Based on a discussion with the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 MSSV manufacturer (i.e., Crosby), HBRSEP, Unit No. 2 has not experienced more problems with the Crosby MSSVs than the nuclear industry in general, thus, the new level of safety will be equivalent to that of the nuclear industry. Second, if a MSSV does fail the surveillance test, the proposed TS will require additional MSSVs to be tested. This requirement provides assurance that testing will reveal possible generic problems. The impact of the tolerance on the Chapter 15 accidents was analyzed and found to be within acceptable limits.

Since no Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis is significantly impacted by the proposed changes, there would be no increase in the consequences of an accident previously evaluated. The testing in accordance with the ASME Boiler and Pressure Vessel Code will provide an adequate level of assurance that the MSSVs will be able to perform their intended function; therefore the probability of a previously evaluated accident is not increased.

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

No new systems or equipment are involved with the proposed changes; and the plant's configuration and operational procedures are unaffected. Since the proposed changes do not impact the plant's operation, it can not create a new or different kind of accident.

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

The change in testing frequency is in a less conservative direction, but it is based on the ASME Code and the improved Standard Technical Specifications. Since HBRSEP, Unit No. 2 has not experienced a greater number of failures associated with these MSSVs than the nuclear industry in general, the decrease in the MSSV testing frequency will not significantly impact the margin of safety. Also, analyses have been performed that demonstrate that the impact of the setpoint tolerance change on the UFSAR Chapter 15 accident analysis results is not significant. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

NRC Project Director: David B. Matthews.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: January 31, 1996.

Description of amendment request: The proposed change would revise the Technical Specifications section 4.4 to allow the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing. A new TS section 6.12 is proposed to describe the containment leakage rate testing program, committing to meet 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B for type A tests; and to meet 10 CFR part 50, Appendix J, Option A, for types B and C tests. The bases would be changed to reflect the proposed changes.

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

The proposed change does not involve a significant hazards consideration for the following reasons.

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

The HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2 Type A testing history provides substantial justification for the proposed test schedule change to one test in a 10 year period. Three Structural Integrity Tests (SITs) and seven Integrated Leak Rate Tests (ILRTs) have been performed with acceptable results. Previous 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.

Therefore, this proposed change to the TS that revises the Type A testing frequency does not involve an increase in the probability of an accident previously evaluated.

This proposed change to revise the test schedule frequency does not impact nor 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 systems and components that penetrate the containment structure.

NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' 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. Section 10.1.2 of NUREG–1493, ''Summary of Technical Findings, Leakage-Testing Intervals,'' states the following.

1. Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks found by Type A tests have been only marginally above existing requirements.

2. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between ILRTs is possible with minimal impact on public risk.

Therefore, based on the previous Type A test results, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change only incorporates the performance based testing approach authorized in 10 CFR 50, Appendix J, Option B, and is 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 this proposed change 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 new or different type of accidents from those previously evaluated.

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

The proposed change does not change the allowable leak rate from the containment, it only allows an extension of the interval between the performance of Type A leak rate testing. NUREG–1493, which 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. Section 10.1.2 of NUREG–1493, ''Summary of Technical Findings, Leakage-Testing Intervals,'' states the following.

''1. Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks found by Type A tests have been only marginally above existing requirements.

2. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between ILRTs is possible with minimal impact on public risk.''

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: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

NRC Project Director: David B. Matthews.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: January 29, 1996.

Description of amendment request: The proposed change would revise the technical specifications (TS) to: (1) add TS 4.6.1.5 to provide criteria for 24-hour full-load testing of the emergency diesel generators (EDGs) to be performed during each refueling outage; (2) revise TS 4.6.1.2 to allow testing of the EDG protective bypasses listed in TS 3.7.1.d to be done independent of the safety injection or loss of offsite power testing; and (3) revise TS 4.6.1.3 to include the EDG protective bypass inspection and a requirement to inspect the EDGs at least once every refueling outage.

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

The proposed changes do not involve a significant hazards consideration for the following reasons.

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

The proposed changes do not involve a significant increase in the probability of an accident previously evaluated. The proposed changes require additional testing of the EDGs and will change the requirement for when the protective bypasses are tested. The function of the EDGs remains unchanged. Since the additional testing involves the EDGs, which are required to mitigate an accident and are not involved in the initiation of an accident, the proposed changes will not increase the probability of an accident.

The proposed changes do not involve a significant increase in the consequences of an accident previously evaluated. The proposed changes require additional testing to verify the reliability of the EDGs and to show the EDGs can withstand maximum accident loading conditions. The proposed changes will also require the testing of the EDG protective bypasses to be accomplished during EDG outages and not during the SI/LOOP testing during a refueling outage. The ability of the EDGs to perform their accident mitigation function remains unchanged. Therefore, the proposed changes will not increase the consequences of an accident.

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

The proposed changes do not create the possibility of a new kind of accident from any previously evaluated. The proposed changes are an enhancement to the EDG testing requirements. The most significant change will require additional testing of the EDGs to demonstrate adequate reliability and to determine if the EDGs can withstand maximum accident loading conditions. The remaining changes will augment the TS to allow on-line EDG inspections and testing. Since the function of the EDGs remains unchanged and they are not the initiator of an accident, the proposed changes will not create the possibility of a new kind of accident from any previously evaluated.

The proposed changes do not create the possibility of a different kind of accident from any accident previously evaluated. The proposed changes require additional testing of the EDGs (i.e., the 24 hour full-load test) and revise the requirement for testing the EDG protective bypasses during the SI/LOOP testing. The additional testing of the EDGs will demonstrate sufficient reliability and determine if the EDGs can withstand maximum accident loading conditions. The EDG protective bypasses will be statically tested during an EDG outage thus preventing possible damage to equipment from a transient if the protective bypass fails. The function of the EDGs remains unchanged by these proposed changes. Since the EDGs are required to mitigate an accident and are not the initiators of an accident, the proposed changes will not create a different kind of accident from any kind of accident previously evaluated.

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

The proposed changes do not reduce the margin of safety as defined in the TS. The proposed changes are being submitted as an enhancement to the testing requirements outlined in the TS. The changes include additional testing, revising the requirement to test the engine protective bypasses during the SI/LOOP testing and clarification of the periodicity of inspecting the EDGs. The additional testing demonstrates increased reliability and determines that the EDGs can cope with maximum accident loading. The remaining proposed changes provide clarification as to when the EDG inspections and testing are required. The ability of the EDGs to perform their function will not be reduced. Therefore, the margin of safety will not be reduced 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: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

NRC Project Director: David B. Matthews.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: December 6, 1995.

Description of amendment request: The proposed amendment would change the technical specifications of these plants to incorporate 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors'', Option 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:

ComEd proposes to revise Byron Nuclear Power Station, Units 1 and 2 (Byron), and Braidwood Nuclear Power Station, Units 1 and 2 (Braidwood) Technical Specification (TS) Section 3/4.6.1, ''Primary Containment,'' and the associated Bases to reflect recent changes to Appendix J to 10 CFR 50, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' The proposed revisions include:

1. Adding TS Definitions 1.15.a for the maximum allowable primary containment leakage rate (L_a) and 1.20.a for the maximum calculated primary containment pressure (P_a). The redundant definitions throughout TS Section 3/4.6.1 are deleted,

2. Adding numerous statements throughout TS Section 3/4.6.1 that leak rate testing is performed in accordance with Regulatory Guide (RG) 1.163, Revision 0, ''Performance-Based Containment Leak-Test Program,'' and its referenced documents,

3. Deleting TS requirements that are taken verbatim from 10 CFR 50, Appendix J. The specific requirements will be placed in the containment leakage rate test program in accordance with RG 1.163, and its referenced documents, and

4. Clarifying Technical Specification Surveillance Requirement (TSSR) 4.6.1.1.a for consistency with NUREG–1431, Revision 1, ''Standard Technical Specifications for Westinghouse Plants.''

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

10 CFR 50, Appendix J, has been amended to include provisions regarding performance-based leakage testing requirements (Option B). Option B allows plants with satisfactory Integrated Leak Rate Testing (ILRT) performance history to reduce the Type A testing frequency from three tests in ten years to one test in ten years. For Type B and Type C tests, Option B allows plants to reduce testing frequency based on the leak rate test history of each component. In addition, Option B establishes controls to ensure continued satisfactory performance of the affected penetrations during the extended testing interval. To be consistent with the requirements of Option B to 10 CFR 50, Appendix J, ComEd proposes to include appropriate changes to the TSs that incorporate the necessary revisions.

Some of the proposed changes represent minor curtailments to current TS requirements, but are based on the requirements specified by Option B to 10 CFR 50, Appendix J. Any such changes are consistent with the current plant safety analyses and have been determined to represent sufficient requirements for the assurance of the reliability of equipment assumed to operate in the safety analyses, or provide continued assurance that specified parameters associated with containment integrity remain within their acceptance limits. The other proposed changes maintain consistency with those requirements specified by Option B to 10 CFR 50, Appendix J and are consistent with the current plant safety analyses. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems affecting the leak rate integrity are not assumed in any safety analyses to initiate any accident sequence; therefore, the probability of occurrence of any accident previously evaluated is not increased. In addition, the proposed changes to the limiting conditions for operation and surveillance requirements for such systems are consistent with the current 10 CFR 50, Appendix J, requirements. The proposed changes maintain an equivalent level of reliability and availability for all affected systems.

Maintaining allowable leakage within the analyzed limit assumed for the accident analyses does not adversely affect either the onsite or offsite dose consequences. Furthermore, containment leakage is not an accident initiator. As such, there is no adverse impact on the probability of accident initiators. Thus, there is no significant increase in the probability or occurrence of any previously analyzed accident, or increase the consequences of any previously analyzed accident.

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

Option B of 10 CFR 50, Appendix J, specifies, in part, that a Type A test may be conducted at a periodic interval based on the performance of the overall containment system. Type A tests measure both the containment system overall integrated leakage rate at the containment pressure boundary and system alignments assumed during a large break loss-of-coolant accident (LOCA), and demonstrate the capability of the primary containment to withstand an internal pressure load. The acceptable leakage rates are specified in the TSs. For Type B and C tests, intervals are proposed for establishment based on the performance history of each component. Acceptance criteria for each component are based upon demonstration that the leakage rates at design basis pressure conditions for applicable penetrations are within the limits specified in the TSs.

The proposed changes reflect the requirements specified in the amended 10 CFR 50, Appendix J, and are consistent with the current plant safety analyses. Some minor curtailments of current TS requirements are based on generic guidance or similarly approved provisions for other plants. These changes do not involve revisions to the design of the plant. Some of the changes may involve revision in the testing of components at the plant; however, these are in accordance with the current plant safety analyses and provide for appropriate testing or surveillance that is consistent with Option B to 10 CFR 50, Appendix J. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current plant safety analyses.

No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect corresponding changes associated with Option B to 10 CFR 50, Appendix J. The proposed changes maintain at least the present level of operability of any such system that affects plant containment integrity. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. The associated systems that affect plant leak rate integrity related to the proposed amendment are not assumed to initiate any accident sequence. In addition, the proposed surveillance requirements for any such affected systems are consistent with the current requirements specified within the TSs and are consistent with the requirements of Option B to 10 CFR 50, Appendix J. The proposed surveillance requirements maintain an equivalent level of reliability and availability of all affected systems and, therefore, do not affect the consequences of any previously evaluated accident. As such, the probability of systems associated with leak rate test integrity failing to perform their intended function is unaffected by the proposed limiting conditions for operation and surveillance requirements.

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

The provisions specified in Option B to 10 CFR 50 Appendix J, allows changes to Type A, B, and C test intervals based upon the performance of past leak rate tests. The effect of extending containment leak rate test intervals is a corresponding increase in the likelihood of containment leakage. The degree to which intervals can be extended has a direct impact on the potential effect on existing plant safety margins and the public health and safety that can occur due to an increased likelihood of containment leakage.

Changing Type A, B, and C test intervals from those currently provided in the TS to those provided for in 10 CFR 50, Appendix J, Option B, slightly increases the risk associated with Type A, B, and C specific accident sequences. Historical data suggest that increasing the Type C test interval can slightly increase the associated risk; however, this is compensated by the corresponding risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. In addition, when considering the total integrated risk, which includes all analyzed accident sequences, the additional risk associated with increasing test intervals is negligible.

The proposed changes are consistent with those provisions specified in Option B of 10 CFR 50, Appendix J, and are consistent with current plant safety analyses. In addition, these proposed changes do not involve revisions to the design of the plant. As such, the proposed individual changes will maintain the same level of reliability of the equipment associated with containment integrity, assumed to operate in the plant safety analysis, or provide continued assurance that specified parameters affecting plant leak rate integrity, will remain within their acceptance limits. Therefore, the proposed changes provide continued assurance of the leakage integrity of the containment without adversely affecting the public health and safety and, as such, will not significantly reduce existing plant safety margins.

The proposed changes are based on United States Nuclear Regulatory Commission (USNRC) accepted provisions and maintain necessary levels of system or component reliability affecting plant containment integrity. The performance-based approach to leakage rate testing concludes that the impact on public health and safety due to revised testing intervals is negligible. The proposed changes will not reduce the availability of systems associated with containment integrity when they are required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment requests: December 4, 19, 19, 20, 20, and 20, 1995.

Description of amendment request: Each proposed amendment would change the surveillance requirement frequency from the current once per 18-month interval to once per 24-month which is the proposed length of a Haddam Neck refueling cycle. The changes pertain to the following equipment:

December 4, 1995, Reactivity control systems flow paths, rod position indication system, and Rod drop time.

December 19, 1995, Containment Air Recirculation System.

December 19, 1995, Main steam line (MSL) Code Safety Valves self actuation, auxiliary feedwater system, service water system, snubber testing, feedwater isolation valve actuation, and primary auxiliary building cleanup system.

December 20, 1995, reactor coolant system (RCS) interlock, containment sump, High Pressure Safety Injection Pump and Low Pressure Safety Injection autostart and alignment, containment spray, and PH control.

December 20, 1995, Trip actuating devices and channel trips, reactor trip system, reactor trip system instrumentation, and accident monitoring instrumentation.

December 20, 1995, RCS flow indicators, Loop stop valve interlock, Pressurizer code safety valves, Emergency power supply for the pressurizer heaters, Containment main sump and volume control tank (VCT) level monitoring system, RCS pressure boundary valves, Low temperature overpressure protection (LTOP) system, and RCS vent path.

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

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

The proposed changes to surveillance requirements of the Haddam Neck Plant Technical Specifications extend the frequency for checking the operability of the affected components/equipment. The proposal would extend the frequency from at least once per 18 months to at least once each refueling interval (i.e., nominal 24-months).

Changing the frequency of surveillance requirements from at least once per 18 months to at least once each refueling interval does not change the basis for the frequency. The frequency was chosen because of the need to perform this verification under the conditions that apply during a plant outage, and to avoid the potential of an unplanned transient if the surveillance were conducted with the plant at power.

The proposed changes do not alter the intent or method by which the surveillance are conducted, do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated. As such, the proposed changes in the frequency of surveillance requirements will not degrade the ability of the equipment/components to perform its safety function.

Additional assurance of the operability of the components/ equipment is provided by additional surveillance requirements (e.g., monthly or quarterly surveillance).

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of surveillance requirements. This evaluation included a review of surveillance results, preventive maintenance records, and the frequency and type of corrective maintenance. It concluded that there is no indication that the proposed extension could cause deterioration in the condition or performance of any of the subject components.

In addition to the substantive changes, there are format changes which are merely editorial and because format changes produce no physical change they do not influence the probability or consequences of accidents.

Since the proposed changes only affect the surveillance frequency for safety systems that are used to mitigate accidents, the changes cannot affect the probability of any previously analyzed accident. While the proposed changes can lengthen the intervals between surveillance, the increases in intervals has been evaluated and it is concluded that there is no significant impact on the reliability or availability of the safety system and consequently, there is no impact on the consequences on any analyzed accident.

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

The proposed changes to surveillance requirements of the Haddam Neck Plant Technical Specifications extend the frequency for verifying the operability of the affected components/equipment. The proposal would extend the frequency from at least once per 18 months to at least once each refueling interval (nominal 24 months).

Changing the frequency of surveillance requirements from at least once per 18 months to at least once each refueling interval does not change the basis for the frequency. The frequency was chosen because of the need to perform this verification under the conditions that apply during a plant outage, and to avoid the potential of an unplanned transient if the surveillance were conducted with the plant at power.

In addition to the substantive changes, there are format changes which are merely editorial and because format changes produce no physical change they do not influence the probability of new or different types of accidents.

The proposed changes do not alter the intent or method by which the surveillance are conducted, do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated. As such, the proposed changes cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes to surveillance requirements of the Haddam Neck Plant Technical Specifications extend the frequency for verifying the operability of the components/equipment. The proposal would extend the frequency from at least once per 18-months to at least once each refueling interval (24-months).

In addition to the substantive changes, there are format changes which are merely editorial and because format changes produce no physical change they do not influence the margin of safety.

The proposed changes to surveillance frequency are still consistent with the basis for the frequency, and the intent or method of performing the surveillance is unchanged. Further, the current inservice testing requirements and the previous history of reliability of the system provides assurance that the changes will not affect the reliability of the auxiliary feedwater system. Thus, it is concluded that there is no impact on the margin of safety.

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

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

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

NRC Project Director: Phillip F. McKee.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County and Northeast Nuclear Energy Company, et al., Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut

Date of amendment request: June 6, 1995 (published August 2, 1995, 60 FR 39434), as supplemented November 22, 1995.

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

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

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

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

The plant operations review committee (PORC) is an oversight group and helps to ensure that the units are operated in a safe manner. To accomplish this the PORCs provide their recommendations on the safety related activities to the Vice President—Haddam Neck Plant for Haddam Neck and to the respective Nuclear Unit Directors for Millstone. Each Millstone Unit has its own PORC. It is proposed that the members of the Millstone PORCs be selected by the respective Nuclear Unit Director based on their knowledge and expertise in specific key plant functions. The Millstone Station has one site operations review committee (SORC). The SORC is also an oversight group whose charter is to advise the Senior Vice President—Millstone Station on all matters related to nuclear safety at the Millstone site. The Haddam Neck Plant, being a single unit site, has one PORC, which advises the Vice President—Haddam Neck Plant. The members of the Haddam Neck Plant PORC will be selected by the Vice President—Haddam Neck Plant based on their knowledge and expertise in specific key plant functions. The PORC and SORC add to the defense-in-depth concept provided by the design, operation, maintenance, and quality oversight by promoting excellence through the conduct of their affairs and by maintaining a diligent watch over their responsibilities.

These administrative changes will revise the composition section of the technical specifications for the PORC members. Millstone Unit individuals will be appointed by the Nuclear Unit Directors if the individual meets one or more of the following areas of expertise: Plant Operations, Engineering, Reactor Engineering, Maintenance, Instrumentation and Controls, Health Physics, Chemistry, Work Planning and Control, and Quality Services. The Haddam Neck Plant, due to its broader scope of review also include an individual experienced in Security and specific expertise in Electrical Maintenance and Mechanical Maintenance. The individuals who will serve on PORC shall continue to meet the criteria of ANSI N18.1–1971 along with the qualification requirements contained in the technical specifications. This approach is consistent with the standard technical specifications and NUREG 0800, Section 13.4. For SORC at the Millstone Station, the method of identifying who shall serve as Vice Chairperson has been modified for clarity. Finally, the individual who shall represent Quality and Assessment Services shall be modified to allow a qualified member of Quality and Assessment Services to serve on SORC.

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

These modifications broaden the unit committee participation and reflect current organizational positions and will not increase the probability of occurrence or the consequences of an accident previously evaluated.

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

The proposed administrative enhancements to the composition of the PORC and Millstone Station SORC will not affect the way in which the units are physically operated. These administrative changes to PORC and SORC continue to meet the guidelines of ANSI N18.7–1976. The modifications to PORC and SORC continue to allow these groups to provide a thorough review of activities at the units.

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

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

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

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, for the Haddam Neck Plant, and the Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, for Millstone 1, 2, and 3.

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

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: November 22, 1995 (NRC–95–0124).

Description of amendment request: The proposed amendment would modify the allowed out-of-service time for one onsite alternating current (ac) electrical power division from 72 hours to 7 days. The proposed amendment would also eliminate accelerated testing and special reports as a result of diesel generator surveillance failures in accordance with Generic Letter 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators,'' dated May 31, 1994.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident. Changing the out-of-service time, surveillance frequency and reporting requirements for emergency diesel generators (EDGs) will not affect the initiation of an accident, since EDGs are not associated with any accident initiation mechanism. The proposed changes will not impact the plant design or method of EDG operation. The increased out-of-service time has been evaluated to have only a small impact on plant risk. Performing the EDG inspections during plant operations will decrease plant risk during plant outages. Deleting the accelerated testing provisions will not affect the consequences of an accident since the implementation of a maintenance and monitoring program for EDGs consistent with the provisions of the maintenance rule will assure EDG performance as discussed in Generic Letter 94–01. Deleting reporting requirements has no impact on consequences of an accident since reporting has no accident effect. Based on the amount of electrical system redundancy, the small increase in plant risk during operations and the decrease in plant risk during outages, this change will not result in a significant increase in the probability or consequences of an accident.

2. The proposed changes do not create the possibility of a new or different accident from any previously evaluated. The proposed changes do not modify the plant design or method of diesel operation. Therefore, no new accident initiator is introduced, nor is a new type of failure created. For these reasons, no new or different type of accident is created by these changes.

3. The proposed changes do not involve a significant reduction in a margin of safety. Since implementation of a maintenance program for the EDGs consistent with the Maintenance Rule will ensure that high EDG performance standards are maintained, the accelerated testing schedule is not needed to maintain the margin of safety. Deleting reporting requirements has no impact on safety or margin of safety. Increasing the allowed out-of-service time for one division of onsite AC power will slightly increase EDG unavailability during plant operation. However, this change does not impact the redundancy of offsite power supplies, the allowed out-of-service time if both divisions are inoperable, or the ability to cope with a station blackout event. This request also does not change the Action statement for AC electrical power systems required when the plant is shutdown. The increase in core damage frequency was assessed to be small by an evaluation using the plant PSA [probabilistic safety assessment] for the operating condition. Enabling the diesel generator inspections to be performed on-line will improve safety while shutdown by reducing EDG out-of-service time during outages. For these reasons, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: John N. Hannon.

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

Date of amendment request: December 21, 1995 (NRC–95–0133).

Description of amendment request: The proposed amendment would implement Option B of the recently revised 10 CFR Part 50 Appendix J in a manner consistent with Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' and industry guidance contained in NEI 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,'' with the exception of previously approved exemptions which the licensee wishes to remain in effect. The previously approved exemptions are for reduced pressure for testing MSIVs [main steam isolation valves] and testing of LPCI [low pressure coolant injection] isolation valves in accordance with Technical Specification (TS) 4.4.3.2.2.

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

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

The proposed change implements the new Option B of 10 CFR Part 50 Appendix J on performance-based containment leakage testing. The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameters or conditions that contribute to the initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the containment structure designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures which result from any loss-of-coolant accident (LOCA). The containment is designed to limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design, only the frequency of measuring Type A, B, and C leakage, the proposed change does not directly result in an increase in containment leakage. However, decreasing the test frequency can increase the probability that an increase in containment leakage could go undetected for an extended period of time. Test intervals will be established based on the performance history of components being tested. The risk resulting from the proposed changes is characterized as follows, based primarily on the results contained in NUREG–1493 [''Performance-Based Containment Leakage Test Program''], the principal Technical Support Document used by the NRC as the basis for the Appendix J final rule (Reference 9 [of application]) and the NRC's Final Regulatory Impact Analysis as contained in SECY–95–181 [Final Regulatory Impact Analysis, Performance-Based Containment Leakage-Test Program (Attachment 2 to NRC Rulemaking Issue Affirmation, SECY–95–181 dated July 17, 1995, Final Amendment to 10 CFR 50, Appendix J, ''Containment Leakage Testing,'' to Adopt Performance-Oriented and Risk-Based Approaches)] (Reference 10 [of application]):

Type A Testing

NUREG–1493 found that the effect of containment leakage on overall accident risk is minimal since risk is dominated by accident sequences that result in failure or bypass of the containment.

Industry wide, ILRTs [integrated leak rate tests] have only found a small fraction of the leaks that exceed current acceptance criteria. Only three percent of all leaks are detectable only by ILRTs, and therefore, by extending the Type A testing intervals, only three percent of all leaks have a potential for remaining undetected for longer periods of time. In addition, when leakage has been detected by ILRTs, the leakage rate has been only marginally above existing requirements. The Fermi Type A testing confirms the industry-wide experience that a majority of the leakage experienced during Type A testing is through components tested by Type B and C tests.

NUREG–1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, show that increasing the Type A leakage test intervals would have a minimal impact on public risk.

Type B and C Testing

NUREG–1493 found that while Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths, performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. The risk model used in NUREG–1493 suggests that the number of components tested would be reduced by about 60 percent with less than a three-fold increase in the incremental risk due to containment leakage. Since, under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small. In addition, the NRC's Final Regulatory Impact Analysis concluded that while the extended testing intervals for Type B and C tests led to minor increases in potential offsite dose consequences, the beneficial expected decrease in onsite (LLRT [local leak rate testing] & ILRT worker) dose exceeds (by at least an order of magnitude) the potential off-site dose consequences.

The editorial change to the bases has no impact on the probability or consequence of an accident since it is strictly a correction to achieve consistency between the bases and the specifications.

Based on the above, DECO [the licensee] has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change involves the reduction of Type A, B, and C test frequency. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequency has no influence on, nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

The editorial change to the bases has no effect on any kind of accident since it is strictly a correction to achieve consistency between the bases and the specifications.

Based on the above, DECO has concluded that the proposed change will not create the possibility [of] a new or different kind of accident previously evaluated.

3. The request does not involve a significant reduction in a margin to safety.

The proposed change only affects the frequency of Type A, B, and C testing. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. However, the proposed change can increase the probability that an increase in leakage could go undetected for an extended period of time. NUREG–1493 has determined that, under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a, which is defined by the Fermi 2 Technical Specifications to be 0.5 percent by weight of the containment air per 24 hours at 56.5 psig (P_a). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin to safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed Technical Specification change.

Except for the method of defining the test frequency, no change in the method of testing is being proposed. The Type B and C tests will continue to be done at full pressure (P_a) or greater with the exception of the Main Steam Isolation Valves, which have an approved exemption. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

The editorial change to the bases has no effect on the margin of safety since it is strictly an editorial change to achieve consistency between the bases and the specifications.

As a result, DECO has concluded that the proposed change will not result in 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: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: John N. Hannon.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: January 22, 1996.

Description of amendment request: The proposed amendment would modify the steam generator tube plugging criteria in Technical Specification 3/4.4.5, Steam Generators, and the allowable leakage in Technical Specification 3/4.4.6.2, Operational Leakage, and the associated Bases. The amendment would allow the implementation of alternate steam generator tube plugging criteria for the tube support plate (TSP)/tube intersections for Unit 1.

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

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

Structural Considerations

Industry testing of model boiler and operating plant tube specimens for free span tubing at room temperature conditions show typical burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements at or below the structural limit of 4.0 volts. One model boiler specimen with a voltage amplitude of 19 volts also exhibited a burst pressure greater than 5000 psi. Burst testing performed on one intersection pulled from STP Unit 1 in 1993 with a 0.51 volt indication yielded a measured burst pressure of 8900 psi at room temperature. Burst testing performed on another intersection pulled from STP Unit 1 in 1995 with a 0.48 volt indication yielded a measured burst pressure of 9950 psi at room temperature.

The projected end-of-cycle (EOC) voltage compares favorably with the 4.7 volt structural limit considering the EPRI [Electric Power Research Institute] voltage growth rate for indications at STP. Using the methodology of the NRC Generic Letter 95–05, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning-of-cycle (BOC) repair limit which should preclude EOC indications from growing in excess of the structural limit. The non-destructive examination (NDE) uncertainty to be applied per EPRI is approximately 20 percent. The EPRI recommended growth allowance of 30 percent/EFPY [effective full power year] is also to be applied. This growth value is conservative for STP Unit 1 based on previous inspection history. By adding NDE uncertainty allowances and a crack growth allowance to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 4.7 volts can be represented as:

RL + (0.20 x RL) + (0.45* x RL) = 4.7 volts, which yields RL of 2.85 volts.

* The 30% growth rate for 1 EFPY was scaled up to the cycle length used at South Texas.

This repair limit (2.85 volts) reasonably could be applied for APC [alternate plugging criteria] implementation to repair bobbin indications greater than the 1.0 volt criterion specified by NRC Generic Letter 95–05 and is independent of RPC [rotating pancake coil-probe] confirmation of the indications. STP has chosen to use a steam generator tube upper repair limit of 2.85 volts to assess tube integrity for those bobbin indications which are above 1.0 volt but do not have confirming RPC calls. This 2.85 volt upper limit for non-confirmed RPC calls is consistent with the NRC Generic Letter 95–05. Since the upper bound for repair of non-confirmed RPC is limited to a value far less than the structural limit associated with a full alternate criteria, the establishment of the repair limits are determined to be reasonable and conservative with respect to the industry pulled tube data base used.

Leakage Considerations

As part of the implementation of APC, the distribution of EOC cracking indications at the TSP intersections has been used to calculate the primary-to-secondary leakage which is bounded by the maximum leakage required to remain within applicable dose limits. This limit was calculated using the Technical Specification RCS [reactor coolant system] Iodine-131 transient spiking values consistent with NUREG–0800. Application of the APC criteria requires the projection of postulated MSLB [main steam line break] leakage based on the projected EOC voltage distribution for the beginning of cycle. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Draft NUREG–1477 requires that all indications to which APC is applied must be included in the leakage projection.

The projected MSLB leakage rate calculation methodology prescribed in EPRI TR–100407 will be used to calculate the EOC leakage. A Monte Carlo approach will be used to determine the EOC leakage, accounting for all of the ECT [eddy current testing] uncertainties, voltage growth, and an assumed probability of detection (POD) of 0.6 for a 1.0 volt repair limit. The fitted logarithmic function probability of leakage correlation will be used to establish the STP MSLB leak rate used for comparison with a bounding allowable leak rate in the faulted loop which would result in radiological consequences which are within applicable dose limits. Due to the relatively low voltage levels of indications at STP and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit.

Therefore, implementation of APC does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences. The proposed amendment does not result in any 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?

Implementation of the proposed steam generator tube alternate plugging criteria for ODSCC [outer diameter stress corrosion cracking] at the TSP intersections does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations since no ODSCC has been identified outside the thickness of the TSPs. It is therefore expected that for all plant conditions, neither a single or multiple tube rupture event would occur in a steam generator where APC has been applied.

Specifically, STP will implement, for Unit 1, a maximum leakage rate of 150 gpd [gallons per day] per steam generator (SG) to help preclude the potential for excessive leakage during all plant conditions. The current technical specification limits on primary-to-secondary leakage at operating conditions are 1 gpm [gallon per minute] for all steam generators or 500 gpd for any one SG. The RG [Regulatory Guide] 1.121 criterion for establishing operational leakage rate limits governing plant shutdown is based upon leak-before-break (LBB) considerations to detect a free span crack before potential tube rupture as a result of faulted plant conditions. The 150 gpd limit is intended to provide for leakage detection and plant shutdown in the event of an unexpected crack propagation resulting in excessive leakage. RG 1.121 acceptance criteria for establishing operating leakage limits are based on LBB considerations such that plant shutdown is initiated if the permissible crack is exceeded.

The predicted EOC leakage for STP is based on the calculated growth rate and does not take credit for the TSP proximity during normal operation. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during the secondary side blowdown of a MSLB. Typically, it is expected for the vast majority of intersections that only partial uncovery will occur. Thus, the proximity of the TSP will enhance the burst capacity of the tube.

Steam generator tube integrity is continually maintained through inservice inspection and primary-to-secondary leakage monitoring. Any tubes falling outside the APC repair limits are removed from service. Therefore, the possibility of a new or different kind of accident from any accident previously developed is not created.

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

The use of the voltage based bobbin probe for dispositioning ODSCC degraded tubes within TSP intersections by APC is demonstrated to maintain steam generator tube integrity in accordance with the requirements of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs [General Design Criterion] 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking are removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevation is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case at STP as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting secondary-to-primary pressure differential on the deformed tubes may cause some of the tube to collapse.

There are two concerns associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that through wall cracks in tubes could sufficiently enlarge during tube deformation or collapse, causing sufficient in-leakage of secondary water back to the core which dilutes the poisoning effect of boron injection from the emergency cooling system. Again, an increase in core PCT may result.

Consequently, since the LBB methodology is applicable to the STP reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The analysis identified tubes located adjacent to wedge regions that are subject to potential collapse during combined LOCA and SSE. These tubes will be excluded from application of APC. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced.

Implementation practices using the bobbin probe voltage based tube plugging criteria bounds RG 1.83 considerations by:

(1) Using enhanced eddy current inspection guidelines consistent with those used by EPRI in developing the correlations. This provides consistency in voltage normalization,

(2) Performing a 100 percent bobbin coil inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length, and

(3) Incorporating RPC inspection for all tubes with larger indications left in service. This further establishes the principal degradation morphology as ODSCC.

Implementation of APC at TSP intersections will decrease the number of tubes which must be repaired. Since the installation of tube plugs (to remove ODSCC degraded tubes from service) reduces the RCS flow margin, APC implementation will help preserve the margin of flow that would otherwise be reduced.

For each cycle the projected EOC primary-to-secondary leak rate allowed is bounded by a leak rate which limits the radiological consequences of a EOC MSLB to within applicable dose limits. Therefore, this change does not involve a significant reduction in the margin to safety.

It is therefore concluded that the proposed license amendment request does not result in a significant reduction in the margin of safety as defined in the plant Final Safety Analysis Report or Technical Specifications.

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, TX 77488.

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

NRC Project Director: William D. Beckner.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: January 22, 1996.

Description of amendment request: The proposed amendment would modify the steam generator tube plugging criteria in Technical Specification 3/4.4.5, Steam Generators, and the associated Bases, to allow the implementation of alternate steam generator tube plugging criteria for the tube-to-tubesheet joints (known in the industry as F*) for Unit 1.

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

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

The proposed changes to the Steam Generator section of Technical Specifications do not affect any accident initiators or precursors and do not alter the design assumptions for the systems or components used to mitigate the consequences of an accident. The requirements approved by the NRC will not be reduced by this request. Since F* utilizes the ''as rolled'' tube configuration that exists as part of the original steam generator design, all of the design and operating characteristics of the steam generator and connected systems are preserved. The F* joint has been analyzed and tested for design, operating and faulted condition loadings in accordance with Regulatory Guide 1.121 safety factors. At worst case, a tube leak would occur with the result being a primary to secondary leak.

Should a tube leak occur, the impact is bounded by the ruptured tube evaluation submitted by STP for the Unit 1 operating license. No new or unreviewed accident conditions are created by the use of F* criteria. The potential for a tube rupture is not increased from the original submittal, thus there is no impact on accidents evaluated as the design basis. Therefore use of the F* criteria will not increase the probability of occurrence of an accident previously evaluated.

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

The use of the proposed F* alternate plugging criteria will not introduce significant or adverse changes to the plant design basis. The failure of a tube which remained unplugged in accordance with the F* criteria would result in a tube leak, which is a previously analyzed condition. Since this leak would occur below the secondary face of the tubesheet, its leak rate would be limited by the tube-to-tubesheet interface. Qualification testing and previous experience indicates that normal and faulted leakage would be well below the technical specification limits creating no threat associated with tube rupture type leakages. This conclusion is consistent with previous F* programs approved and used at other operating plants.

However, in the unlikely event the failed tube severed completely at a point below the F* region, the remaining F* joint would retain engagement in the tubesheet due to its length of expanded contact within the tubesheet bore, preventing any interaction with neighboring tubes. If the tube severs at a point above the F* region, then it is covered by the tube rupture event as a part of the UFSAR [Updated Final Safety Analysis Report]. Thus, the possibility of a new or different type of accident from any accident previously evaluated is not created.

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

Based on previous responses (above), the protective boundaries of the steam generator are preserved. A tube with degradation can be kept in service through F* criteria which provided an un-degraded expanded interface with the tubesheet and which satisfies all of the necessary structural and leakage requirements in accordance with Regulatory Guide 1.121 and the Technical Specifications. Since the joint is constrained within the tubesheet bore there is no additional risk associated with tube rupture. Since the UFSAR analyzed accident scenarios remain bounding, the use of an F* criteria does not reduce the margin of safety.

Thus, these changes do not involve a significant reduction in the margin of safety. Therefore, based on the above evaluation, STP has concluded that these changes do not involve any significant hazards considerations.

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, TX 77488.

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

NRC Project Director: William D. Beckner.

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

Date of amendment requests: January 12, 1996 (AEP:NRC:1233).

Description of amendment requests: The proposed amendments would modify technical specification section 4.4.11 to eliminate the surveillance requirement (SR) demonstrating operability of the emergency power supply for the pressurizer power-operated relief valves (PORVs) and block valves.

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:

Per 10 CFR 50.92, a proposed change does not involve significant hazards consideration if the change does not:

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

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.

Criterion 1

The proposed change is consistent with NUREG–1431 [Standard Technical Specifications Westinghouse Plants]. Due to the high reliability and continued testing of the Class 1E power supply, we conclude that the elimination of the SR will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed change does not involve the addition of any new plant operation or procedures, and the elimination of the SR is consistent with NUREG–1431. For these reasons, we believe that the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The proposed change is consistent with NUREG–1431, and it does not affect the acceptance criteria of any of the other PORV and block valve tests currently performed. For these reasons, we believe that the proposed amendment will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and the applicable Bases of the Standard Technical Specifications Westinghouse Plants. The Bases for the applicable surveillance, 3.4.11.4, states ''This Surveillance is not required for plants with permanent 1E power supplies to the valves.'' Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: January 22, 1996.

Description of amendment request: The proposed change relocates the containment isolation valve (CIV) list, Table 3.6–2, from the Technical Specifications to the Technical Requirements Manual (TRM). This change affects Technical Specifications Sections 1.8.1a, 4.6.1.1a, 3.6.3.1, 4.6.3.1.1 and 4.6.3.1.2, and the Basis Section 3/4.6.3. A note at the bottom of Table 3.6–2 regarding the CIVs that are subject to administrative control is retained in the Technical Specifications by relocating it to Sections 1.8.1a and 4.6.1.1a. This change is being performed in accordance with Generic Letter 91–08, which provides guidance for removal of component lists from the Technical Specifications.

Additionally, a change to provide relief in the surveillance requirement in Section 4.6.1.1a is included. The change allows valves, blind flanges, and deactivated automatic valves located inside the containment and are locked, sealed, or otherwise secured in the closed position to be verified closed during each cold shutdown but not more often than once per 92 days. The current requirements check the valve position once per 31 days.

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

Pursuant to 10CFR50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed changes. NNECO concludes that these changes do not involve a significant hazards consideration (SHC) since the proposed changes satisfy the criteria in 10CFR50.92(c). That is, the proposed changes do not:

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

The proposed change to remove the Containment Isolation Valve (CIV) list from the Technical Specifications will not result in any hardware or operating changes. The proposed change is based upon NRC Generic Letter (GL) 91–08 and merely removes the CIV table and all references to the table from the technical specifications without affecting the operability requirements of any of the listed valves. The technical specifications will continue to require the CIVs to be operable. Limiting Condition for Operation and surveillance requirements for the valves will also remain in the technical specifications. The CIV table will be relocated to the Millstone Unit No. 2 Technical Requirements Manual (TRM) which is controlled in accordance with 10CFR50.59.

This change is administrative in nature and does not involve an increase in the probability or consequence of an accident previously evaluated. Furthermore, the proposed change does not alter the design, function, or operation of the valves involved, and therefore does not affect the probability or consequences of any previously evaluated accident.

The change to Section 4.6.1.1a that reduces the surveillance requirement for valves, blind flanges, and deactivated automatic valves located inside the containment provides consistency with NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants'' as well as the Technical Specifications of Millstone Unit No. 3, Haddam Neck Plant, and Seabrook. The probability or consequences of any previously evaluated accidents are not affected.

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

The change to relocate the CIV list from the technical specifications to the TRM will not impose any different operational or surveillance requirements, nor will the change remove any such requirements. Adequate control of information will be maintained. Furthermore, as stated above, the proposed change does not alter the design, function, or operation of the valves involved, and therefore no new accident scenarios are created.

The change to Section 4.6.1.1a that reduces the surveillance requirement for valves, blind flanges, and deactivated automatic valves located inside the containment does not alter the design, function, or operation of the valves involved, and therefore no new accident scenarios are created.

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

The proposed changes will not reduce the margin of safety since it has no impact on any safety analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor does the proposed change affect any instrument setpoints or equipment safety functions.

The relocation of the valve list is consistent with the guidance provided in GL 91–08. The change to the surveillance interval is consistent with NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants'' as well as the Technical Specifications of Millstone Unit No. 3, Haddam Neck Plant, and Seabrook. The intent of the technical specification will be met since the change will not alter function or operability requirements for any CIV.

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

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

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

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: January 17, 1996.

Description of amendment request: The amendment request would delete a license requirement to submit responses to and to implement requirements of Generic Letter 83–28, because the requirement has been completed. Generic Letter 83–28 pertains to the Salem anticipated transient without scram (ATWS) event.

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 an SHC because the change would not:

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

NNECO's proposal to delete License Condition 2.C(4) is an administrative change. The NRC Staff has accepted Millstone Unit No. 3's responses regarding the actions required by GL 83–28, thus, the license condition has been met and is no longer necessary. The proposed change does not affect the configuration, operation, or performance of any system, structure, or component. Additionally, the limiting conditions for operation, limiting safety system settings, and safety limits specified in the Millstone Unit No. 3 Technical Specifications are unchanged. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The NRC Staff has accepted Millstone Unit No. 3's responses regarding the actions required by GL 83–28, thus, the license condition has been met and is no longer necessary. The proposed change to delete License Condition 2.C(4) does not affect the configuration, operation, or performance of any system, structure, or component. Additionally, the limiting conditions for operation, limiting safety system settings, and safety limits specified in the Millstone Unit No. 3 Technical Specifications are unchanged. Therefore, this proposed change cannot create the possibility of a new or different kind of accident from any previously analyzed.

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

The NRC Staff has accepted Millstone Unit No. 3's responses regarding the actions required by GL 83–28, thus, the license condition has been met and is no longer necessary. The proposed change to delete License Condition 2.C(4) does not affect the configuration, operation, or performance of any system, structure, or component. Additionally, the limiting conditions for operation, limiting safety system settings, and safety limits specified in the Millstone Unit No. 3 Technical Specifications are unchanged. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

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

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

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

Date of amendment request: December 22, 1995.

Description of amendment request: The proposed changes will revise Limerick Generating Station, Units 1 and 2, Technical Specification 3.6.1.8 ''Drywell and Suppression Chamber Purge System,'' increasing the Drywell and Suppression Chamber Purge System operating time limit from 90 hours each 365 days to 180 hours each 365 days.

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

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

These TS changes do not increase the probability of occurrence of an accident previously evaluated in the SAR [Safety Analysis Report]. This activity involves changing the allowable operating limit for the Drywell and Suppression Chamber Purge System from 90 hours each 365 days to 180 hours each 365 days. This change increases the probability that this system will be in service should a LOCA [loss of coolant accident] occur, but does not increase the probability that a LOCA will occur.

Increasing the operating limit for the Drywell and Suppression Chamber Purge System from 90 hours to 180 hours each 365 days does not increase the consequences of a LOCA as previously evaluated in the SAR. These proposed TS changes increase the probability of a LOCA occurring during the time the Drywell and Suppression Chamber Purge System is in operation, and therefore, increase the probability of the failure of the operating SGTS [Standby Gas Treatment System] filter bank. However, the risk to containment integrity was previously evaluated and found to be acceptable (UFSAR [Updated Final Safety Analysis Report] Section 9.4.5.1.2.2 and WASH—1400 ''Reactor Safety Study'').

Increasing the duration that the vent/purge line isolation valves may be open does not increase the probability that these valves will not perform as designed (i.e., close upon receipt of an isolation signal) in response to a LOCA. However, the changes will increase the likelihood that the vent and purge valves will be called on to close. As discussed in UFSAR Section 6.2.4.2, the containment purge valves have undergone extensive testing and analyses to demonstrate the operability of these valves following a LOCA.

In addition to the existing Safety Analysis Report (SAR) evaluations, a Level 2 PSA [Probabilistic Safety Assessment] Analysis (containment failure) was performed to determine the additional risk associated with changing the operating limit from 90 to 180 hours each 365 days. The PSA evaluation conservatively assumed a 200 hour vent/purge duration per a 365 day period. The figure of merit evaluated is the large early release frequency (LERF) which represents the likelihood of containment failure following core damage that could significantly affect the public (e.g., release of a large amount of radioactive material early enough in the accident that evacuation of the public has not occurred). The 200 hour vent/purge duration increased the LERF approximately 3% from the base value of 2.57E–8 for all PSA initiators. This analysis concluded that the increase in risk of containment failure is well within the bounds of the EPRI [Electrical Power Research Institute] PSA Applications Guideline for permanent changes. The same relative increase applies to the large Design Basis Accident LOCA LERF.

These changes do not directly or indirectly degrade the performance of any other safety systems (assumed to function in the accident analysis) below their design basis. The potential for other equipment failures in the reactor enclosure due to duct-work impact, impingement, and the resulting environmental conditions was evaluated. It was concluded that the environmental qualifications for the LGS equipment are sufficient to ensure operability under the predicted environmental conditions, and there is no impact or impingement-related damage to essential equipment. Although the probability of occurrence of a malfunction of equipment important to safety is increased, the existing SAR analysis and Level 2 PSA Analysis demonstrate the increased risk and radiological consequences are not significant.

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

This activity does not change the function of the Drywell and Suppression Chamber Purge System, the containment isolation system, or SGTS as previously evaluated in the SAR. Changing the duration of operation of the vent and purge system does not create an accident initiator not considered in the SAR. Therefore, the possibility of an accident of a different type is not created.

This activity does not create a failure mode not considered in the SAR. All possible equipment failures that could occur as a result of a LOCA during high volume purging have previously been identified and evaluated in the SAR. Therefore, this activity does not create the possibility of a different type of malfunction of equipment important to safety.

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

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

The Bases of Technical Specification 3.6.1.8 states that the intent of the 90 hour per 365 day operating limit for the Drywell and Suppression Chamber Purge System is to protect the integrity of the SGTS filters. As discussed above, the requirements specified in ODCM paragraph 3.3.6 assure the availability of the backup SGTS filter train during operation of the vent and purge system. Furthermore, as discussed above, revising the operating limit from 90 hours to 180 hours each 365 days does not involve a significant increase in risk. The margin of safety as defined in the Bases of Technical Specification 3.6.1.8 is maintained.

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

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 6, 1996.

Description of amendment request: The amendments would change the Technical Specifications to lower the 125 Volt Battery Charger surveillance amperage from at least 200 amps to at least 170 amps.

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

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

The proposed amendment will permit replacement of aging battery chargers while ensuring these replacement battery chargers will restore the battery from the design minimum charge to its fully charged state while supplying normal steady-state loads. This meets the design basis for the 125V DC system and is consistent with Salem Unit 1 and 2 commitment to IEEE 308–1971 in UFSAR Section 3A.

The 125V DC battery chargers are not addressed as a contributor to any accident analyzed in the UFSAR, therefore, changes to the battery charger output current will not increase the probability of an accident occurring.

The limiting analyzed accident considered in this proposed TS amendment is the Loss of Offsite Power coincident with a Loss of Coolant Accident. This is currently the limiting design duty cycle for the batteries. The 125V batteries are sized to maintain all emergency loads for a period of 2 hours without battery chargers. This is demonstrated by performing the surveillance specified in TS 4.8.2.3.2.f, which is not being changed. Since the chargers are not required to be available during this 2 hour period, and since the proposed charging rate will supply the necessary loads following restoration of AC power, the proposed amendment will have no effect on the consequences of this accident.

The current limiter is calculated to extend the recharging time from 20 hours to 30 hours, but this is not considered significant since two, sequential battery discharge events are not considered plausible.

PSE&G calculation substantiates the capability of the chargers to restore the battery from the design minimum charge to its fully charged state while supplying normal steady-state loads following a Station Blackout (SBO) Event which exceeds the current design duty cycle.

In addition, a review of 125V DC Battery System load profiles indicated that the battery chargers are capable of supplying expected loads when restoring the battery from a design minimum charge state to a fully charged state irrespective of the status of the plant.

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

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

The proposed amendment does not result in any design or physical configuration changes to the 125V DC system. This change supports the installation of the replacement chargers and ensures the chargers are surveilled within the bounds of limiting input amperage. No changes are being made to the function, design basis, or operation of the 125V DC system by this proposed change. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment to TS 4.8.2.3.2.e ensures that the replacement battery chargers have sufficient capacity to restore each 125V battery from the design minimum charge to its fully charged state while supplying normal steady-state loads. A margin of safety is maintained on both the AC input and DC output of the chargers since the specified current is above that required to support the 125V DC system and will result in AC current below the ampacity rating of the battery charger input cables.

Testing to a charger output current of at least 170 amps will maintain a margin of safety to the current required during actual worst case normal loading on the 125V DC buses.

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

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 9, 1996.

Description of amendment request: The proposed amendment would allow an installed overhead door assembly, to be used in lieu of the equipment hatch closure, to isolate the hatch opening to the containment building during fuel movement and core alterations.

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 Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. Containment closure is used with respect to the mitigation of fuel handling accidents, and as such, any change to these requirements will not affect the probability of an accident. The proposed changes will also not result in a significant increase in the consequences of an accident previously analyzed since the technical specification requirements remain bounded by the fuel handling accident assumption of no containment closure.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. Containment closure is not assumed in the accident analyses for Ginna Station. Also, the proposed change remains acceptable with respect to SRP [NUREG–800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, July 1981''] 15.7.4 and GDC [General Design Criterion] 19 requirements. Therefore, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: February 9, 1996.

Description of amendment request: The proposed amendment would incorporate the methodology for determining the Low Temperature Overpressure Protection (LTOP) limits into the Administrative Controls Section 5.6.6 of the Ginna Technical Specifications (TS). The proposed amendment will allow the licensee to perform future LTOP evaluations, using NRC-approved methodology, without requiring changes to the TS.

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

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes only require that future LTOP limits be developed using NRC approved methodology as specified within the Administrative Controls section and do not involve any technical changes. As such, these changes are administrative in nature and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because

the changes do not impact any safety analysis assumptions other than requiring future evaluations of LTOP limits to be performed in accordance with NRC approved methodology. These changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: February 9, 1996.

Description of amendment request: The proposed amendment would revise the Technical Specifications setpoints for steam generator (SG) water level-high feedwater isolation function. It would take advantage of a greater allowable operating band for SG water level afforded by replacement SGs.

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 Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed setpoint change does not degrade the performance of any plant equipment. Therefore, the probability of an accident is not increased. Since the revised trip setpoint and allowable value remain bounded by the accident analysis value of 100% steam generator narrow range level, the consequences of any accident are not adversely affected.

2. Operation of Ginna Station in accordance with the proposed changes 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 physical alteration to the plant (i.e., no new or different types of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The revised setpoint and allowable value remain bounded by the accident analysis assumptions. The existing values are based on design considerations and not accident analysis parameters. The replacement steam generators are not restricted by the same design considerations with respect to the ESFAS [engineered safety features actuation system] Steam Generator Water Level—High function. Therefore, this change does not involve a reduction in a margin of safety.

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

Local Public Document Room Location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: February 9, 1996.

Description of amendment request: The proposed amendment would change Technical Specification 5.3.1 to allow the use of Zirlo fuel cladding material.

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 methodologies used in the accident analysis remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore accident analyses are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design bases.

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5 fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO clad rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. Since the original design criteria are met, the ZIRLO clad fuel rods will not be an initiator for any new accident. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO clad fuel rods. Since the dose predictions in the safety analyses are not sensitive to fuel rod cladding material, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

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

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

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for other VANTAGE 5 fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The ZIRLO cladding material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems or components are not changed in any manner, nor is the reliability of any structure, system or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems or components are not affected, 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.

Use of ZIRLO cladding material does not change the VANTAGE 5 reload design and safety limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, the fuel assemblies will be evaluated using NRC-approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

The use of Zircaloy-4, ZIRLO or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC-approved methodologies will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC-approved methods that have been approved for application to the fuel configuration.

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: January 30, 1996.

Description of amendment request: The proposed amendments would modify the Technical Specifications to increase the minimum allowable reactor coolant system total flow rate from 284,000 gpm (for Unit 1) and 275,300 gpm (for Unit 2) to 295,000 gpm for both units. Through the 1980's and into the 1990's the North Anna Unit 1 and 2 steam generators experienced increasing levels of steam generator tube plugging. There was a corresponding decrease in the reactor coolant flow rate. As a result, the Commission issued several amendments in the 1989 to 1992 time frame to reduce the minimum reactor coolant flow rate. Subsequently, the licensee replaced the steam generators in both units, with steam generators having an increased number of tubes compared to the replaced steam generators. With the increased number of tubes and less flow resistance, a greater reactor coolant flow rate is attainable. When the amendments were issued decreasing the minimum required reactor coolant flow rate, the transmittal letters stated the revision was temporary and would be increased when the steam generators were replaced.

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 probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report would not increase. The proposed Technical Specifications change only increases the minimum allowable RCS total flow rate in the applicable Limiting Condition of Operation. No other changes are being made to allowable operating conditions defined by Technical Specifications, procedures, or to any plant design feature by the implementation of this change. There is no impact on the actual plant performance. Changes in the assumed initial conditions for the accident have no bearing on the probability of occurrence of the assumed accident or malfunction. The RCS flow rate is an assumption in applicable safety analyses. Existing analyses of record have assumed RCS flow rates which are bounding with respect to expected actual plant behavior. Therefore, the implementation of the proposed Technical Specifications change does not affect the probability nor increase the consequences of an accident previously evaluated.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report would not be created. The proposed change to North Anna Units 1 and 2 Technical Specifications Table 3.2–1 does not involve any alterations to the physical plant which would introduce any new or unique operational modes or accident precursors. Only the allowable value for measured Reactor Coolant System Total Flow Rate will be changed.

3. The margin of safety as defined in the basis for any technical specifications is not reduced. The proposed Technical Specifications change only increases the minimum allowable RCS total flow rate in the applicable Limiting Condition of Operation. The RCS flow rate is an assumption in applicable safety analyses. Existing analyses of record have assumed RCS flow rates which are bounding with respect to expected actual plant behavior. Therefore, the margin of safety is not reduced by the proposed increase in the allowable RCS Total Flow Rate.

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

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

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

NRC Project Director: David B. Matthews.

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

Date of amendment request: January 31, 1996.

Description of amendment request: The amendments would revise the Technical Specifications to reduce the minimum volume of fuel that must be maintained in the diesel generator day tanks from 750 to 450 gallons. The amendments would also revise the surveillance requirements for the diesel generators to permit some surveillances to be performed while the reactor units are at power where the licensee considers it safe to do so without compromising the availability of the diesel generators to perform their intended function.

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 an increase in the probability of occurrence of an accident previously evaluated.

The proposed changes do not result in any physical modifications to any plant systems or components nor change the operation of any plant equipment. The EDG [emergency diesel generator] fuel oil supply system will continue to provide adequate fuel supply to the EDGs in a manner consistent with applicable accident analyses. Performing surveillance tests or portions of surveillance tests at power that do not jeopardize stable plant operations does not increase the probability of occurrence of previously analyzed accidents.

Therefore, there is no increase in the probability of occurrence of any accident.

2. Increase the consequences of an accident previously evaluated.

The proposed changes do not result in any physical modifications to any plant systems or components nor change the operation of any plant equipment. The EDG fuel oil system remains capable of supplying the EDGs with sufficient quantities of fuel oil to provide power for long term loss of offsite power. The EDG surveillances will continue to be performed in a manner that will ensure that the EDGs will be capable of performing their intended safety functions. The proposed changes to the electrical distribution system surveillances will continue to ensure that the electrical distribution system remains operable to power the required safety systems.

Therefore, these proposed changes will not result in an increase in the consequences of any evaluated accidents.

3. Create the possibility for an accident of a different type than was previously evaluated.

The proposed changes do not result in any physical modifications to any plant systems or components nor change the operation of any plant equipment. Only those surveillance tests or portions of surveillance tests that do not jeopardize stable plant operation will be performed at power. Overlap testing to fully test the electrical distribution system protection functions does not introduce any unique accident precursors. The EDG fuel oil system remains capable of supplying the EDGs with sufficient quantities of fuel oil to provide power for long term loss of offsite power. The EDG surveillances will continue to be performed in a manner that will ensure that the EDGs will be capable of performing their intended safety functions.

Therefore, there are no new precursors generated that would result in the possibility of a different type of an accident than was previously evaluated in the SAR [Safety Analysis Report].

4. Decrease the margin of safety as described in the bases section of Technical Specifications.

The EDG fuel oil system will continue to provide adequate fuel supply in a manner consistent with applicable accident analyses. The EDG surveillances will continue to be performed in a manner that will ensure that the EDGs are capable of performing their intended safety functions. The proposed changes to the electrical distribution system surveillances will continue to ensure that the electrical distribution system remains operable to power the required safety systems.

Therefore, the margin of safety as described in the Technical Specifications is not reduced.

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

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

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

NRC Project Director: David B. Matthews

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

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

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

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

Date of application for amendment: July 21, 1995, August 8, 1995, and December 15, 1995.

Brief description of amendment request: The proposed amendment would modify the requirements for testing an emergency diesel generator (EDG) when the other is inoperable. The amendment would correct an editorial error in the Duane Arnold Energy Center Operating License and would correct an erroneous reference in the Technical Specification.

Date of publication of individual notice in Federal Register: February 2, 1996 (61 FR 3953).

Expiration date of individual notice: March 4, 1996.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Nuclear Power Plant, Unit No. 1, San Luis Obispo County, California

Date of amendment request: January 18, 1995.

Description of amendment request: The proposed amendment would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, to allow operation of Unit 1 in Mode 3 (Hot Standby) during replacement of nonvital auxiliary transformer 1–1. Specifically, TS 3/4.8.1.1, ''Electrical Power Systems—A.C. Sources—Operating,'' Action Statement (a), would be revised to permit a one-time extension of the allowed outage time (AOT) from 72 hours to 120 hours.

Date of individual notice in Federal Register: February 1, 1996 (61 FR 3737).

Expiration of individual notice: March 4, 1996.

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

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

Date of amendment request: February 5, 1996, as supplemented by letter dated February 14, 1996.

Brief description of amendment request: The amendment changes Technical Specifications 4.6.2.3.b, ''Suppression Pool Cooling'', and TS 4.6.2.2.b, ''Suppression Pool Spray'', to include flow through the RHR heat exchanger bypass line (in addition to the RHR heat exchanger) in the Suppression Pool Cooling and Suppression Pool Spray flow path used during RHR pump testing.

Date of publication of individual notice in Federal Register: February 9, 1996 (61 FR 5040).

Expiration date of individual notice: March 11, 1996.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

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

Date of application for amendment: January 16, 1996.

Brief description of amendment request: The proposed amendment would change the Technical Specification surveillance frequency for the drywell bypass leakage rate test from 18 months to 120 months (10 years) with a more frequent testing requirement if performance degrades. Additionally, specific leakage limits would be deleted for the air lock seal and barrel tests. Also, surveillance frequencies for the air lock interlock test and seal pneumatic system leak test would be changed from 18 months to 24 months. Finally, the surveillance frequencies for the air lock barrel test would be changed from ''each COLD SHUTDOWN if not performed within the previous 6 months'' to ''at least once per 24 months'' and from 18 months to 24 months. The licensee requested that this amendment be approved for use during the current refueling outage which began on January 27, 1996.

Date of publication of individual notice in Federal Register: February 2, 1996 (61 FR 3951).

Expiration date of individual notice: March 4, 1996.

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

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.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendments: December 7, 1995.

Brief description of amendments: The amendments add the convolution analytical technique for the analysis of the pre-trip main steam line break event to the list of approved core operating limits analytical methods listed in Technical Specification 6.9.1.9, ''Core Operating Limits Report.'' The convolution analytical technique was previously reviewed and approved by the NRC staff and the supporting safety evaluation was provided to Baltimore Gas and Electric Company by letter dated May 11, 1995.

Date of issuance: February 5, 1996.

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

Amendment Nos.: 210 and 188.

Facility Operating License No. DPR–53 and DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 177)

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated February 5, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 2, 1994, as supplemented by letters dated November 16 and December 14, 1995.

Brief description of amendments: The amendments delete the content of the Appendix B, ''Environmental Protection Plan'' (Non-radiological) Technical Specifications and modify License Condition 2.C.(2) so as to delete that portion which refers to the Environmental Protection Plan.

Date of issuance: February 5, 1996.

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

Amendment Nos.: Unit 1–164—Unit 2–146.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications and License Conditions.

Date of initial notice in Federal Register: March 1, 1995 (60 FR 11131). The November 16 and December 14, 1995, letters provided clarifying information that did not change the scope of the November 2, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 5, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

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

Date of application for amendments: January 13, 1995, as supplemented by letter dated August 30, 1995.

Brief description of amendments: The amendments revise the Technical Specifications to increase the surveillance test intervals and allowed outage times for the Reactor Trip System and Engineered Safety Features Actuation System. The NRC staff has reviewed the proposed changes and finds that, with one exception as noted in the enclosed Safety Evaluation, the amendments conform to WCAP–10271, ''Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation Systems,'' with its revisions and supplements, provides appropriate limiting conditions for operation and action statements, and is, therefore acceptable.

Date of issuance: February 16, 1996.

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

Amendment Nos.: Unit 1–165—Unit 2–147.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 15, 1995 (60 FR 14019).

The August 30, 1995, letter provided clarifying information that did not change the scope of the January 13, 1995, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 16, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

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

Date of application for amendments: July 10, 1995.

Brief description of amendments: These amendments modify the Technical Specifications to minimize the potential for boron dilution of the reactor coolant system (RCS) during startup of an isolated RCS loop. The changes permit RCS loop isolation only during Modes 5 and 6 and require the RCS loop isolation valves be open with power removed from their valve operators during Modes 1, 2, 3, and 4. The changes also require isolation of primary grade water from the RCS during Modes 4, 5, and 6, except during planned boron dilution or makeup activities.

Date of issuance: February 12, 1996.

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

Amendment Nos.: 195 and 78.

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

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42602).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 20, 1995, as supplemented December 4, 1995.

Brief description of amendments: These amendments revise Technical Specification 3/4.8.1.1, ''A.C. Sources-Operating,'' to incorporate guidance provided in NRC Generic Letter (GL) 84–15, ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,'' and GL 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' which includes (1) revised requirements for testing the operable emergency diesel generators (EDGs) for various combinations of inoperable offsite circuits and EDGs and (2) revised surveillance requirements for the EDGs. The revised surveillance requirements include specifying generator voltage, frequency limits, and diesel starting time. The amendments also make several editorial changes to TS 3/4.8.1.1 to make TS 3/4 8.1.1 consistent with the guidance provided in the NRC's Improved Standard Technical Specifications (NUREG–1431).

Date of issuance: February 12, 1996.

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

Amendment Nos.: 196 and 79.

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

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42603).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 22, 1995.

Brief description of amendments: The amendments consist of changes relating to removal of the TS Bases from the TS index.

Date of issuance: February 13, 1996.

Effective date: February 13, 1996.

Amendment Nos.: 182 and 176.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65678).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 31, 1995, as supplemented November 28, 1995, and December 21, 1995. The supplementary submittals did not affect the staff's proposed finding of no significant hazards consideration.

Brief description of amendment: This amendment increases the surveillance interval on various instruments from 18 to 24 months.

Date of issuance: February 13, 1996.

Effective date: February 13, 1996.

Amendment No.: 152.

Facility Operating License No. DPR–72. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35070).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: October 16, 1995, as supplemented by letter dated December 22, 1995.

Brief description of amendments: The amendments add a footnote to Technical Specification 4.6.1.2.d stating the Type B and C tests scheduled for Unit 1's refueling outage, cycle 6 (1R6) will be conducted in accordance with Option B of 10 CFR Part 50, Appendix J (hereafter referred to as Option B) using the guidance of Regulatory Guide 1.163, September 1995. This change only applies to Unit 1's refueling outage 1R6 because implementation of Option B for Type A, B, and C testing for both units is being incorporated into the Improved TS that are scheduled to become effective after refueling outage 1R6.

Date of issuance: February 2, 1996.

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

Amendment Nos.: Unit 1–93—Unit 2–71.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62490).

The December 22, 1995, letter provided clarifying information that did not change the scope of the October 16, 1995, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 2, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

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

Date of application for amendment: June 9, 1995, as supplemented November 9, 1995.

Brief description of amendment: The amendment relocates Surveillance Requirement 4.6.6.1.d.3 to TS 3.6.6.2 and revises the Action Statement of Section 3.6.6.1 to decouple it from Section 3.6.6.2. In addition, Definition 1.12, ''Secondary Containment Boundary'' is deleted and included in the Bases Section 3/4.6.6, Secondary Containment. Bases Section 3/4.6.6.2, Secondary Containment is expanded using the guidance of the improved standard technical specifications (STS) for Westinghouse plants (NUREG–1431).

Date of issuance: February 5, 1996.

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

Amendment No.: 126.

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

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39445).

The November 9, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 17, 1995, as supplemented October 16, 1995, and November 28, 1995.

Brief description of amendments: The amendments revise the Prairie Island Radiological Effluent Technical Specifications and other sections relating to radiological controls to conform to NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1, and Generic Letter 89–01, ''Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.''

Date of issuance: January 24, 1996.

Effective date: January 24, 1996, with full implementation within 120 days.

Amendment Nos.: Unit 1–122; Unit 2–115.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52933).

By letters of October 16, 1995, and November 28, 1995, NSP forwarded a copy of its revised ODCM to the NRC for use as a reference and provided additional clarifying information. This information did not change the licensee's amendment request, the scope of the original Federal Register notice or the staff's initial proposed no significant hazards considerations determination. Therefore, renoticing was not warranted. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 24, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 28, 1995.

Brief description of amendments: The amendment eliminates the Technical Specifications requirements to perform 10 CFR Part 50, Appendix J, Type C hydrostatic tests on certain valves that are assured a water seal following a Design Basis Accident.

Date of issuance: February 8, 1996.

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

Amendment Nos.: 110 and 73.

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

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49941).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 19, 1995, as supplemented December 21, 1995.

Brief description of amendment: The amendment revises Technical Specification Section 2.2, ''Safety Limits,'' to change the minimum critical Power ratio safety Limit due to use of General Electric 13 fuel product line.

Date of issuance: February 8, 1996.

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

Amendment No. 111.

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

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52934).

The December 21, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the Federal Register notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 28, 1995.

Brief description of amendments: The amendments delete the operability and surveillance requirements involving secondary containment differential pressure instrumentation.

Date of issuance: As of date of issuance, to be implmented within 30 days.

Effective date: February 14, 1996.

Amendment Nos.: 112 and 74.

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

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49942).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 14, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 28, 1995.

Brief description of amendments: These amendmends revise Technical Specifications Table 4.3.1.1–1, ''Reactor Protection System Instrumentation Surveillance Requirements,'' to reflect changes the surveillance test frequency requirements for various Reactor Protection System instrumentation.

Date of issuance: February 14, 1996.

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

Amendment Nos.: 113 and 75.

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

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49944).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 14, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 20, 1995, as supplemented by letter dated December 18, 1995.

Brief Description of amendment: The Technical Specification (TS) revision represents changes to TS Section 3/4.11.2.6, ''Explosive Gas Mixture,'' TS Table 3.3.7.11–1, ''Radioactive Gaseous Effluent Monitoring Instrumentation,'' and TS Table 4.3.7.11–1, ''Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements.'' The revision removes these TS from the Technical Specifications and relocates the Bases to the Hope Creek Updated Final Safety Analysis Report and the Surveillance Requirements to the applicable surveillance procedures. The Limiting Conditions for Operation are eliminated.

Date of issuance: February 6, 1996.

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

Amendment No.: 91.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39452)

The December 18, 1995 supplement did not effect the proposed no significant hazards determination, contained in the January 20, 1995 application or the Federal Register notice (60 FR 39452).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 1996

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

Date of application for amendment: October 7, 1995 as supplemented by letter dated October 27, 1995.

Brief description of amendment: This amendment changes Technical Specification (TS) 4.8.1.1.2, ''A.C. Sources—Operating,'' by replacing the reference to an upper voltage and frequency band for the 10-second, Emergency Diesel Generator (EDG), starting time test with a minimum required voltage and frequency that must be attained within 10 seconds. The change to TS 4.8.1.1.2 also includes several related changes to TS 4.8.1.1.2 as follows: (1) the requirement for an EDG to achieve 514 rpm, within 10 seconds following a start signal during testing is eliminated, (2) the term ''standby'' replaces the term ''ambient'' in describing the EDG test restart condition, and (3) the term ''must'' is replaced with the term ''may'' in describing the use of manufacturers recommendations for EDG loading.

Date of issuance: February 6, 1996.

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

Amendment No.: 92.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58405)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

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

Date of application for amendment: May 26, 1995, as supplemented May 5, 1995, and January 26, 1996.

Brief description of amendment: The proposed change was to allow the storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) Uranium-235 (U–235) in the new (fresh) and spent fuel storage racks and change the license to reflect changes related to the nuclear fuel cycle.

Date of issuance: February 6, 1996.

Effective date: February 6, 1996.

Amendment No.: 60.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications and License.

Date of initial notice in Federal Register: September 26, 1995 (60 FR 49636)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

Date of application for amendment: May 26, 1995, as supplemented by letters dated July 17, August 14, August 31, September 18, October 6, October 18, November 1, November 16, two letters of November 20, November 21, November 22, two letters of November 27, November 30, December 8, and December 28, 1995; and November 27, 1995; and May 23, 1994, as supplemented by letters dated June 15, 1994, July 11, July 15, November 1, and November 16, 1995; and September 15, 1992, as supplemented April 20, 1993, April 26, 1995, and July 27, 1995.

Brief description of amendment: (1) a full conversion from the licensee's current Technical Specifications (TSs) to a set of TSs based on NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 0, dated September 1992 (including approved travellers used in the issuance of Revision 1, dated April 1995), in response to the licensee's application dated May 26, 1995, as supplemented by letters dated July 17, August 14, August 31, September 18, October 6, October 18, November 1, November 16, two letters of November 20, November 21, November 22, two letters of November 27, November 30, December 8, and December 28, 1995. (2) a revision to the TSs to implement the amended regulation 10 CFR Part 50, Appendix J, Option B (new rule), to provide a performance based option for leakage-rate testing of containment, in response to the licensee's application dated November 27, 1995. (3) a revision to the TSs regarding allowable primary coolant levels of specific activity, in response to the licensee's application dated May 23, 1994, as supplemented by letters dated June 15, 1994, July 11, July 15, November 1, and November 16, 1995. (4) a revision to the TSs adding new requirements that enhance the reliability of power-operated relief valves and block valves (PORV/BV) along with TS changes that provide additional low-temperature overpressure protection, in response to the licensee's application dated September 15, 1992, as supplemented April 20, 1993, and April 26, 1995. By letter dated July 27, 1995, the licensee withdrew this amendment request; however, the licensee rescinded this withdrawal request by letter dated December 28, 1995. Therefore, the proposed changes to the PORV/BV, as requested in the licensee's letter dated May 26, 1995, as supplemented December 28, 1995, are incorporated into this amendment.

Date of issuance: February 13, 1996.

Effective date: February 13, 1996.

Amendment No.: 61.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications and License.

Date of initial notice in Federal Register: December 8, 1995 (60 FR 63071); September 26, 1995 (60 FR 49636); August 30, 1995 (60 FR 45184); July 6, 1994 (59 FR 34669).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

Date of application for amendments: December 30, 1993, as supplemented by letters dated June 3, 1994, August 25, 1994, January 3, 1995, and January 19, 1995.

Brief description of amendments: The amendments replace, in their entirety, the current technical specifications (TS) with a set of TS based on NUREG–1432, ''Standard Technical Specifications—Combustion Engineering Reactors,'' September 1992.

Date of issuance: February 9, 1996.

Effective date: February 9, 1996, to be implemented by August 9, 1996.

Amendment Nos.: Unit 1—Amendment No. 127; Unit 2—Amendment No. 116.

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

Date of initial notice in Federal Register: September 28, 1994 (59 FR 49434) The January 3, 1995, and January 19, 1995, supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 8, 1995 (TS 95–24).

Brief description of amendments: The amendments implement the change to 10 CFR Part 50, Appendix J to incorporate Option B, a voluntary performance-based option, for determining the frequency for performing Type A, B, and C Containment Leak Rate Testing.

Date of issuance: February 5, 1996.

Effective date: February 5, 1996.

Amendment Nos.: 217 and 207.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 182).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 1996.

No significant hazards consideration comments received: None.

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

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

Date of application for amendments: December 8, 1995 (TS 95–20).

Brief description of amendments: The amendments decrease the frequency for conducting air or smoke tests of the containment spray system headers and Residual Heat Removal System headers from every 5 years to every 10 years to verify each spray nozzle is unobstructed.

Date of issuance: February 7, 1996.

Effective date: February 7, 1996.

Amendment Nos.: 218 and 208.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: January 3, 1996 (61 FR 182).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 7, 1996.

No significant hazards consideration comments received: None.

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

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

Date of application for amendment: November 22, 1993 supplemented May 5 and December 20, 1995.

Brief description of amendment: The amendment revised the Technical Specifications to reflect the replacement of analog temperature instrumentation associated with leak detection with digital equipment.

Date of issuance: January 29, 1996.

Effective date: January 29, 1996, and implemented not later than 120 days following startup from the fifth refueling outage.

Amendment No.: 79.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 12, 1994 (59 FR 24752).

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated January 29, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 2, 1995, supplemented January 26, 1996.

Brief description of amendment: The amendment only revised the containment personnel air lock Technical Specifications and added a license condition to allow the air locks to be open in Modes 4 and 5 during core alterations except for movement of recently irradiated fuel. All other provisions of the request are being deferred for further review.

Date of issuance: February 2, 1996.

Effective date: To be implemented not later than 90 days after issuance.

Amendment No. 80.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications and added a license condition.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62497) The supplemental letter provided clarification of administrative controls that will be in place, did not change the initial no significant hazards consideration determination, and was within the scope of the notice issued December 6, 1995.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 2, 1996.

No significant hazards consideration comments received: No.

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

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: December 30, 1995, as supplemented by letters dated July 28, (TXX–95187), September 14, (TXX–95235), and November 29, 1995 (TXX–95299), and January 2, 1996 (TXX–96–003).

Brief description of amendments: These changes authorized usage of the high density fuel storage racks, to increase the spent fuel storage capacity, and to adopt the wording, content, and format of the Improved Standard Technical Specifications.

Date of issuance: February 9, 1996.

Effective date: February 9, 1996.

Amendment Nos.: Unit 1—Amendment No. 46; Unit 2—Amendment No. 32.

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

Date of initial notice in Federal Register: February 1, 1995 (60 FR 6313).

The additional information contained in the supplemental letters dated July 28, (TXX–95187), September 14, (TXX–95235), and November 29, 1995 (TXX–95299), and January 2, 1996 (TXX–96–003), was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated February 9, 1996, and a Safety Evaluation dated February 9, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 19, 1995.

Brief description of amendments: The amendments increase the surveillance test interval for the turbine reheat stop and intercept valves from at least once per 31 days to at least once per 18 months, extend the visual and surface disassembly inspection interval of the turbine reheat stop and intercept valves to 60 months and revise the inspection criteria for the throttle, governor, reheat stop, and reheat intercept valve disassembly inspections.

Date of issuance: February 8, 1996.

Effective date: February 8, 1996.

Amendment Nos.: 195 and 176.

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

Date of initial notice in Federal Register: October 25, 1995 (60 FR 54725).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 20, 1995, as supplemented January 23, 1996.

Brief description of amendments: The amendments revise the North Anna Units 1 and 2 Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing.

Date of issuance: February 9, 1996.

Effective date: February 9, 1996.

Amendment Nos.: 196 and 177.

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65685). The January 23, 1996 supplement provided clarifying information that was within the scope of the December 20, 1995 notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 9, 1996.

No significant hazards consideration comments received: No.

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

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 March 29, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

Date of application of amendments: February 6, 1996.

Brief description of amendments: The amendments revised Technical Specification Section 3.16, ''Containment Hydrogen Control Systems.'' The change adds a footnote to TS 3.16.3.b. to allow a one-time outage duration extension in regard to the Containment Hydrogen Control System flow path. This extension is necessary to install and test plant modifications, which will allow the Containment Hydrogen Control System to perform as designed, without the potential for inoperability due to water accumulation in the flow path.

Date of Issuance: February 7, 1996.

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

Amendment Nos.: Unit 1–214–Unit 2–214–Unit 3–211.

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

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated February 7, 1996.

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

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street NW., Washington, DC 20036.

NRC Project Director: Herbert N. Berkow.

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

Date of application for amendment: February 10, 1996.

Brief description of amendment: The amendment revises Technical Specifications (TS) Surveillance Requirements 4.7.6.c.2, 4.7.6.d, 4.9.11.b.2 and 4.9.11.c regarding the testing methodology utilized by Virgil C. Summer Nuclear Station, which determines the operability of the charcoal filters in the engineering safety features air handling units.

Date of issuance: February 10, 1996.

Effective date: February 10, 1996.

Amendment No.: 131.

Facility Operating License No. NPF–12: Amendment revises the TS.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration, are contained in a Safety Evaluation dated February 10, 1996.

Public comments requested as to proposed no significant hazards consideration: No.

No significant hazards consideration comments received: None.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Dated at Rockville, Maryland, this 21st day of February 1996.

For the Nuclear Regulatory Commission.

Steven A. Varga,

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

[FR Doc. 96–4342 Filed 2–27–96; 8:45 am]

BILLING CODE 7590–01–P

Availability of Draft Branch Technical Position on the Use of Expert Elicitation in the High-Level Waste Program

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of Availability.

SUMMARY: The Nuclear Regulatory Commission is announcing the availability of the ''Draft Branch Technical Position (BTP) on the Use of Expert Elicitation in the High-Level Waste (HLW) Program.''

DATES: The comment period expires May 14, 1996.

ADDRESSES: Send comments to Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555–0001. ATTENTION: Docketing and Services Branch. Hand deliver comments to 11545 Rockville Pike, Rockville, Maryland 20852–2738, between 7:45 a.m. and 4:15 p.m., on Federal workdays.

A copy of the draft BTP is available for public inspection and/or copying at the NRC Public Document Room, 2120 L Street (Lower Level), NW., Washington, DC 20555–0001. Copies of the draft BTP may also be obtained by contacting Karen S. Vandervort, Mail Stop T–7F3, U.S. Nuclear Regulatory Commission. Telephone: (301) 415–7252.

FOR FURTHER INFORMATION CONTACT: Michael P. Lee, Performance Assessment and High-Level Waste Integration Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission, 11545 Rockville Pike, MD 20852–2738. Telephone: (301) 415–6677.

SUPPLEMENTARY INFORMATION: The U.S. Department of Energy (DOE) is conducting a program of site characterization to gather enough information, about the Yucca Mountain (Nevada) site, to be able to evaluate the waste isolation capabilities of a potential geologic repository. Should the site be found suitable, DOE will apply to the NRC for permission to construct and then operate a proposed geologic repository for the disposal of spent nuclear fuel and other high-level radioactive waste at Yucca Mountain. As with other licensing decisions,