Tuesday, December 19, 1995—8:30 A.M. until 6:00 P.M.

Wednesday, December 20, 1995—8:30 A.M. until 6:00 P.M.

Thursday, December 21, 1995—8:30 A.M. until 6:00 P.M.

During this meeting the Committee plans to consider the following:

A. Review of NRC's Programmatic Approach to Low-Level Waste Management. The Committee will conclude its deliberations and issue a report on the alternatives to the future course of the NRC's Low-Level Radioactive Waste Disposal Program.

B. National Research Council/ National Academy of Science Committee Report on the Technical Bases for Yucca Mountain Standards. The NRC staff will discuss with the Committee its insights on the subject report.

C. International Atomic Energy Agency (IAEA) Activities. The Committee will meet with a representative of the IAEA to discuss relevant waste-related activities.

D. Meeting with the Director, NRC's Division of Waste Management, Office of Nuclear Materials Safety and Safeguards. The Director will discuss items of current interest related to the Division of Waste Management programs. Among the topics to be discussed: pilot test of survey and statistical methodology for site decommissioning, status of HLW program, and public comment on program options for NRC's LLW program.

E. ACNW Priorities. The Committee will review Task Action Plans for the initial grouping of priority review issues identified by the Committee.

F. Committee Activities/Future Agenda. The Committee will consider topics proposed for future consideration by the full Committee and Working Groups. The Committee will also discuss ACNW-related activities of individual members.

G. Miscellaneous. The Committee will discuss miscellaneous matters related to the conduct of Committee activities and organizational activities and complete discussion of matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings were published in the Federal Register on September 27, 1995 (60 FR 49924). In accordance with these procedures, oral or written statements may be presented by members of the public, electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the Chief, Nuclear Waste Branch, Mr. Richard K. Major, as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting may be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief, Nuclear Waste Branch prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. Major if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Richard K. Major, Chief, Nuclear Waste Branch (telephone 301/415–7366), between 8:00 A.M. and 5:00 P.M. EDT.

ACNW meeting notices, meeting transcripts, and letter reports are now available on FedWorld from the ''NRC MAIN MENU.'' Direct Dial Access number to FedWorld is (800) 303–9672; the local direct dial number is 703–321–3339.

The ACNW meeting dates for Calendar Year 1996 are provided below:

ACNW Meeting No. and 1996 ACNW Meeting Dates

81—January 24–26, 1996

82—March 27–29, 1996

83—May 2–4 or May 15–17, 1996 (TBD)

84—June 26–28, 1996

85—August 21–23, 1996

86—September 25–27, 1996

87—October 22–23, 1996

88—December 10–12, 1996

Dated: November 30, 1995.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 95–29661 Filed 12–5–95; 8:45 am]

BILLING CODE 7590–01–P

UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 10, 1995, through November 24, 1995. The last biweekly notice was published on November 27, 1995 (60 FR 58395).

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 January 5, 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: May 2, 1995

Description of amendments request: The proposed change revises the large-break loss-of-coolant accident (LOCA) dose consequences. The large-break LOCA dose calculation is being changed to include an additional release path through allowable steam generator tube leakage to the atmospheric dump valves (ADVs) or turbine bypass valves (TBVs).

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

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

The probability or consequences of an accident previously evaluated are not significantly increased by this change to the large break LOCA dose consequences. This change has no effect on the LOCA safety analysis for emergency core cooling system performance, which demonstrates conformance to the acceptance criteria of 10 CFR 50.46, as described in the PVNGS Updated Final safety Analysis Section 6.3.3. This change has no effect on structures, systems or components prior to a LOCA or any other accident. The new radiological consequences of the revised large break LOCA dose calculation are below 10 CFR 100 limits for the exclusion area boundary (EAB) and low population zone (LPZ), and the 10 CFR 50, Appendix A, GDC 19 limits for the control room, as shown in Table 1–1, Column C. The NRC has previously approved changes to the PVNGS LOCA dose consequences with the acceptance criteria that the doses are still within the guidelines set forth in 10 CFR 100 and GDC 19. This acceptance criteria is described in the Safety Evaluation related to amendment Nos. 64, 50, and 37 to PVNGS Units 1, 2, and 3 respectively, dated September 8, 1992.

The LOCA dose calculation is being changed to include an additional release path through allowable steam generator tube leakage to the ADVs or TBVs. This change is necessary to reflect a revised calculation assumption that, following a large break LOCA, the secondary system pressure would fall below reactor coolant system pressure and containment pressure when operators cooldown the steam generators by using ADVs or the TBVs (in accordance with the safety analysis and EOPs [emergency operating procedures]). It is desirable to use the ADVs or TBVs to vent secondary system steam and thus reduce heat input to the reactor coolant system following a large break LOCA. No other LOCA analysis assumptions are being changed, and no changes are being made to structures, systems, components or procedures.

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

This change has no impact on any structures, systems, components, or procedures. The only impact is the revised radiological consequences of a large break LOCA to include an additional release path, as discussed in the response to Standard 1 above. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change to the large break LOCA dose consequences does not involve a significant reduction in a margin of safety. The new radiological consequences of the revised large break LOCA dose calculation are below 10 CFR 100 limits for the EAB and LPZ, and the 10 CFR 50, Appendix A, GDC 19 limits for the control room, as described in the response to Standard 1 above. The NRC has previously approved changes to the PVNGS LOCA dose consequences with the acceptance criteria that the doses are still within the guidelines set forth in 10 CFR 100 and GDC 19. This acceptance criteria is described in the Safety Evaluation related to amendment Nos. 64, 50, and 37 to PVNGS Units 1, 2, and 3 respectively, dated September 8, 1992. No equipment qualification is affected by the new assumption of a release path through the secondary system following a large break LOCA, and no post LOCA radiation zones will be changed. This change has no impact on any structures, systems, components, or procedures.

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: November 22, 1995

Description of amendment request: The current Technical Specifications (TS) Section 3.3.4.2 describes the limiting condition during which components in the Service Water (SW) system may be inoperable. The TS Section 3.3.4.2 states, in part, ''During power operation, the requirements of 3.3.4.1 may be modified to allow any one of the following components to be inoperable provided the remaining systems are in continuous operation.'' The proposed change will delete the qualifying statement,''... provided the remaining systems are in continuous operation,'' from TS Section 3.3.4.2. Currently, this statement requires the ''remaining systems to be in continuous operation'' while allowing one SW loop header, or one SW pump, or one SW booster pump to be inoperable for a period of 24 hours.

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

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

The proposed change would remove the requirement for the remaining SW system components to be in continuous operation while one TS-required component is inoperable. Rather, the remaining components would remain operable, and no change would be made in normal system operation. The SW system provides an accident mitigation function and is not involved in accident initiation sequences. Therefore, the proposed change would not involve a significant increase in the probability of an accident previously evaluated.

The capacity of the SW system is such that its accident mitigation function can be performed by operation of a maximum of two SW pumps, one SW booster pumps, and one SW header. While a TS-required component is inoperable, sufficient accident mitigation capability is provided by the remaining operable components, rather than requiring the remaining systems to be in continuous operation. Therefore, the proposed change would not cause 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 would remove the requirement for the remaining SW system components to be in continuous operation while one TS-required component is inoperable. Rather, the remaining components would remain operable. The proposed change would not change the normal operation of the system, nor would any physical modifications result from the change. The function and capability of the SW systems would remain unchanged. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would remove the requirement for the remaining SW system components to be in continuous operation while one allowed TS-required component is inoperable. Rather, the remaining TS-required components would remain operable. Adequate assurance of operability is maintained by performance of regular surveillance testing. Maintaining operable status rather than placing equipment in continuous operation does not result in a change in the ability of the SW system to perform its intended function, since the system provides an automatic response to accident conditions, and the system possesses adequate capacity to perform its normal operating function with one allowed TS-required component inoperable. Therefore, the 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: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: David B. Matthews

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

Date of amendment request: October 24, 1995

Description of amendment request: The proposed amendment will increase the trip setpoints and allowable values for the low power block (P–7).

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

In accordance with 10CFR50.92, CYAPCO has reviewed the proposed change and has concluded that it does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed 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 evaluated.

The proposed change will relax the power level values for the P–7 interlock by 2 percent. This change affects both the P–7 and P–7N interlocks. The P–7 interlock affects reactor trips on 1) low flow in more than one reactor coolant loop, 2) reactor coolant pump bus under voltage, 3) more than one reactor coolant pump breaker open, 4) main steam line isolation valve closure, 5) turbine trip, and 6) variable low pressure. The P–7 interlock automatically blocks these reactor trips on decreasing power and automatically unblocks these reactor trips on increasing power. The P–7N interlock affects the reactor trip on wide range, neutron flux, high startup rate. P–7N automatically enables this reactor trip on decreasing power level and automatically blocks this reactor trip on increasing power level. The Applicable Modes requirement and Action Statements for the P–7 interlock and the reactor trips associated with both P–7 and P–7N in the Instrumentation Channel and Surveillance Requirements of Technical Specification 3/4.3.1 are being changed by 2 percent to be consistent with the change to P–7. The interlock setpoint cannot cause an accident. Also, the proposed 2 percent increase in the power level still results in a power level well below the power level at which the P–7 interlocked reactor trips are required for accident mitigation, as well as maintaining the high startup rate trip enabled at a higher power level. This proposed power level is consistent with the technical specification requirement prior to the conversion to standard format technical specifications and is also consistent with the Standard Westinghouse technical specification value. Therefore, the proposed change can neither increase the consequences of the design basis accident nor the probability of occurrence of the design basis accidents.

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

The proposed change only modifies the power level for the P–7 and P–7N interlocks. The proposed setpoint is a power level at which stable plant conditions are easier to maintain while transferring the power supply for the reactor coolant pumps between offsite power and the main generator. The setpoint is also well below the power level for which the reactor protection afforded by the trips that are bypassed by P–7 is needed. This cannot create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change maintains the power level for the P–7 interlock below the power level for which the reactor trips that are blocked by the P–7 interlock are required. It also raises the power level to a value at which it is easier to maintain stable plant conditions. This will reduce the likelihood of an automatic reactor trip during the transferring of power for the reactor coolant pumps between offsite power and the main generator. The proposed change will result in the high startup rate reactor trip being enabled at a higher power level. This is conservative since it expands the range of coverage for the trip. Therefore, the proposed change does not impact the margin of safety.

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

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

Date of amendment request: November 1, 1995

Description of amendment request: The proposed amendment will modifiy Surveillance Requirement 4.6.3.2, ''Containment Isolation Valves,'' (CIVs) to change the surveillance interval from at least once per 18 months to at least once per refueling interval.

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:

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

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications extends the frequency for verifying that each CIV actuates to its required position in response to a safety injection actuation test signal. The proposal would extend the frequency from at least once per 18 months to at least once per refueling interval (24 months + 25% as allowed by Technical Specification 4.0.2).

The proposed change to Surveillance Requirement 4.6.3.2 does not alter the intent or method by which the surveillance is conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated.

Additional assurance of CIV operability is provided by Surveillance Requirement 4.6.3.3. Surveillance Requirement 4.6.3.3 requires the confirmation of the mechanical operability of the CIVs by the inservice inspection program. The proposed change does not modify these requirements.

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of Surveillance Requirement 4.6.3.2. This evaluation included a review of surveillance results, preventive maintenance records, and corrective maintenance records. It has been concluded that the CIVs are highly reliable, and that there is no indication that the proposed extension could cause deterioration in valve condition or performance.

As such, the proposed change to the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the CIVs to perform their safety function.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications extends the frequency for verifying that each CIV actuates to its required position in response to a safety injection actuation test signal. The proposal would extend the frequency from at least once per 18 months to at least once per refueling interval (24 months + 25% as allowed by Technical Specification 4.0.2).

The proposed change does not alter the intent or method by which the surveillance is conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change in the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the CIVs to perform their safety function.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications extends the frequency for verifying that each CIV actuates to its required position in response to a safety injection actuation test signal. The proposal would extend the frequency from at least once per 18 months to at least once per refueling interval (24 months + 25% as allowed by Technical Specification Section 4.0.2).

The proposed change does not alter the intent or method by which the surveillance is conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change in the frequency of Surveillance Requirement 4.6.3.2 will not degrade the ability of the CIVs to perform their safety function.

Additional assurance of the operability of the CIVs is provided by Surveillance Requirement 4.6.3.3.

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of Surveillance Requirement 4.6.3.2. This evaluation included a review of surveillance results, preventive maintenance records, and corrective maintenance records. It has been concluded that the CIVs are highly reliable, and that there is no indication that the proposed extension could cause deterioration in valve condition or performance.

Based on the above, the proposed change to Surveillance Requirement 4.6.3.2 of the Haddam Neck Plant Technical Specifications does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: 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

Consumers Power Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: November 8, 1995, as supplemented November 17, 1995

Description of amendment request: The proposed amendment would remove the prescriptive Type A containment leakage test rate frequency of 40 plus or minus 10 months and add a reference to perform containment leakage rate tests in accordance with the criteria specified in Appendix J of 10 CFR Part 50 as amended by approved exemptions. In addition, the proposed amendment would revise the test pressure for Type B and C testing to correct a typographical error.

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

Leakage test rate frequency

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

This change is administrative in nature and does not impact plant systems, structures or components. The proposed change will allow the facility's technical specifications to be revised to allow containment sphere leakage testing in accordance with Appendix J to 10 CFR Part 50 as modified by approved exemptions.

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

This change is administrative in nature and does not impact plant syst ems, structures or components. The proposed change will allow the facility's technical specifications to be revised to allow containment sphere leakage testing in accordance with Appendix J to 10 CFR Part 50 as modified by approved exemptions.

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

This change is administrative in nature and does not impact plant systems, structures or components. The underlying purpose of Appendix J is still achieved. Appendix J states that the leakage test requirements provide for periodic verification testing of the leak tightness integrity of the primary reactor containment. The appendix further states that the purpose of the tests is to assure that leakage through the primary containment shall not exceed the allowable leakage rate values as specified in the technical specifications or associated bases. As stated previously, for Big Rock Point and a large percentage of other plants, the Appendix J Type B and C testing programs provide the most significant and meaningful assessment of containment leak tightness. The testing history and structural capability of the containment establish that there is significant assurance that the extended interval between Type A tests will not adversely impact the integrity of the containment.

Test pressure revision

As stated in the technical specification change request, this revision is being performed to be consistent with accident pressure, P_a, used for Big Rock Point. 20 psig is a typographical error. 23 psig has always been used for these tests.

The proposed change does not:

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

This change is administrative in nature and does not impact plant systems, structures or components. The proposed change will allow the facility's technical specifications to be revised to reflect current containment sphere leakage testing in accordance with Appendix J to 10 CFR Part 50.

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

This change is administrative in nature and does not impact plant systems, structures or components. The proposed change will allow the facility's technical specifications to be revised to reflect current containment sphere leakage testing in accordance with Appendix J to 10 CFR Part 50.

3) involve a significant reduction in a margin of safety.

This change is administrative in nature and does not impact plant systems, structures or components.

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: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770

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

NRC Project Director: Brian E. Holian, Acting

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 amendment request: October 16, 1995

Description of amendment request: Appendix J of 10 CFR Part 50, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' has recently been revised to include Option B. This option allows the implementation of a performance based Type B and C testing program. The proposed change will add a footnote to Technical Specification (TS) 4.6.1.2.d stating that the Type B and C tests scheduled for Unit 1 refueling outage Cycle 6 (1R6) will be conducted in accordance with Option B and using the guidance of Regulatory Guide 1.163, Revision 0. This option is being incorporated into the licensee's request to implement the improved TS. However, the improved TS are not scheduled to become effective until after the Unit 1 refueling outage 1R6.

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 change does not involve a significant increase in probability or consequences of an accident previously evaluated. The proposed change does not involve a change to structures, systems, or components which would affect the probability or consequences of an accident previously evaluated in the Vogtle Electric Generating Plant (VEGP) Final Safety Analysis Report (FSAR). The proposed change only provides a mechanism within the Technical Specifications for implementing a performance-based method of determining the frequency for leak rate testing which has been approved by the NRC via a revision to 10 CFR 50, Appendix J.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed. The amendment will not change the design, configuration, or method of plant operation. It only allows for the implementation of Option B of 10 CFR 50, Appendix J for Unit 1 refueling outage 1R6 without violating the plant Technical Specifications.

3. Operation of VEGP, Unit 1, in accordance with the proposed change will not involve a significant reduction in the margin of safety. The proposed change does not affect a safety limit, an LCO [limiting condition for operation], or the way plant equipment is operated. The NRC is aware that changes similar to this proposed change are required in order to implement Option B of 10 CFR 50, Appendix J. In fact, the staff indicates in Paragraph V.B. of Appendix J that Option B or parts thereof may be adopted by a licensee 30 days after the rule becomes effective by submitting notification of its implementing plan and a request for revision to Technical Specifications. Since the NRC has approved the provision for performance-based testing and must approve this Technical Specification[] change before the performance-based Option B can be implemented, the margin of safety will not be significantly reduced.

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

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308

NRC Project Director: Herbert N. Berkow

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 30, 1995 (noticed in the Federal Register July 5, 1995, (60 FR 35080) as supplemented by letter dated November 20, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications as follows:

1. The Surveillance Frequency for the drywell bypass test is changed from 18 months to 10 years with an increased testing frequency required if performance degrades.

2. The following changes are requested for the drywell air lock testing: (a) the leakage rate surveillance is moved from the air lock Limiting Condition for Operation (LCO) to the drywell LCO, (b) the requirement for the air lock to meet a specific overall leakage limit is deleted, (c) the Note that an inoperable air lock door does not invalidate the previous air lock leakage test is deleted, (d) the Note which required that the air lock leakage test at 3 psid be preceded by pressurizing the air lock to 19.2 psid is moved to the bases, and (e) the Surveillance Frequency for the air lock leakage test and interlock test is changed from 18 months to 24 months.

3. The Actions Notes in the drywell air lock LCO and the drywell isolation valve LCO that identifies that the Actions required by the drywell LCO must be taken when the drywell bypass leakage limit is not met is deleted.

4. The requirement for the drywell air lock seal leakage rate to meet a specific leakage limit is deleted.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration for River Bend Station (RBS) and Grand Gulf Nuclear Station (GGNS), which is presented below:

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

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in surveillance interval. Each of these types of change are discussed below:

1. The administrative changes clarify the format of the requirement or change therequirement to match the design bases of the plant. Clarifying administrative format of

the Technical Specifications does not result in any changes to the Technical Specification requirements and, as a result, does not involve a significant increase in the probability or consequences of an accident previously evaluated. Also, changing the requirements of the Technical Specifications to more closely match the design bases of the plant will continue to assure that the plant will respond as assumed in the accident analyses and, as a result, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes relocate information to the Technical Specification Bases. In the Technical Specifications Bases the relocated information will be maintained in accordance with 10 CFR 50.59 and subject to the change control provisions in Chapter 5 of Technical Specifications. Since any changes to the Technical Specifications Bases will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

3. The proposed changes in frequency for the drywell bypass leakage and drywell air lock surveillances will continue to ensure that no paths exist through passive drywell boundary components that would permit gross leakage from the drywell to the primary containment air space and result in bypassing the primary containment pressure-suppression feature beyond the design basis limit. The Mark III primary containment system satisfies General Design Criterion 16 of Appendix A to 10 CFR Part 50. Maximum drywell bypass leakage was determined previously by reviewing the full range of postulated primary system break sizes. The limiting case was a primary system small break loss of coolant accident (LOCA) and yielded a design allowable drywell bypass leakage rate limit of approximately 35,000 scfm for GGNS and 46,000 scfm (the Technical Specification limit is based on a lower limit of 40,110 scfm) for RBS. The Technical Specifications acceptable limit for the bypass leakage following a surveillance is less than 10% of this design basis value. The most recent bypass leakage value was approximately 2.5% for GGNS and .91% for RBS of the design allowable leakage rate limit for the limiting event. EOI is committed to maintaining programmatic and oversight controls that ensure that drywell bypass leakage remains a small fraction of the design allowable leakage limit.

The drywell is typically exposed to essentially 0 psig during normal plant operation and 3 psig during drywell bypass leak rate testing. These pressures are considerably lower than the structural integrity test pressure and are less likely to initiate a crack or cause an existing crack to grow. Visual inspections of the accessible drywell surfaces that have been performed since the structural integrity tests have not revealed the presence of additional cracking or other abnormalities. Therefore, additional cracking of the drywell structure is not expected due to testing or operation and, similar to the justification for the ten year 10 CFR 50 Appendix J Type A test interval, it is not considered credible for the passive drywell structure to begin to leak sufficiently to impact the design drywell bypass leakage limit.

The primary containment's ability to perform its safety function is fairly insensitive to the amount of drywell leakage, thereby providing a margin to loss of the drywell safety function that is not normally available for safety systems. This insensitivity is demonstrated by the extremely high limiting event design basis allowable leakage for the drywell (e.g., 35,000 scfm for GGNS and 46,000 scfm for RBS). The limiting leakage is almost an order of magnitude higher for other events. Additionally, an even higher allowable leakage can be realistically accommodated by the primary containment due to the margins in the containment design. Because of the margins available, it will take valves in multiple penetration flow paths leaking excessively to cause the primary containment to fail as a result of overpressurization, the probability that drywell isolation valve leakage will result in primary containment failure due to excessive drywell leakage is not considered significant and this drywell/primary containment failure mode is not considered credible.

The proposed Technical Specification changes have no significant impact on the GGNS Individual Plant Examination (IPE) or the RBS IPE conducted per NRC Generic Letter 88–20. The IPEs considered overpressurization failure of primary containment as part of the primary containment performance assessment. Due to the magnitude of acceptable drywell leakage and the extremely low probabilities of achieving such leakage, primary containment failure due to preexisting excessive drywell leakage was considered a non significant contributor to primary containment failure. Primary containment overpressurization failure can occur with or without preexisting excessive drywell leakage in a severe accident. This is due to physical phenomena associated with potentially extreme environmental conditions inside primary containment following a severe accident. However, the calculated frequency of such extreme conditions is very small. The proposed changes do not impact the IPE evaluated phenomena causing primary containment overpressurization failure nor significantly increase the probability that the drywell has preexisting excessive leakage and therefore would not contribute to these accident scenarios.

For the reasons discussed above, the proposed changes do not have any significant risk impact to accidents previously evaluated and do not significantly increase the consequences of an accident previously evaluated. Additionally, drywell leakage is not the initiator of any accident evaluated; therefore, changes in the frequency of the surveillance for drywell leakage does not increase the probability of any accident evaluated.

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

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

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in surveillance interval. Each of these types of change are discussed below:

1. The administrative changes in the Technical Specification requirements do not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change 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.

2. The proposed relocation of requirements does not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements. Adequate control of the information will be maintained in the Technical Specification Bases. Thus, the change proposed does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change modifies the surveillance frequency for drywell bypass leakage and drywell air lock surveillances. The changes only impact the test frequency and do not result in any change in the response of the equipment to an accident. The changes do not alter equipment design or capabilities. The changes do not present any new or additional failure mechanisms. The drywell is passive in nature and the surveillance will continue to verify that its integrity has not deteriorated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in surveillance interval. Each of these types of changes are discussed below:

1. The administrative changes in the Technical Specification requirements do not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. Thus, this change does not cause a significant reduction in the margin of safety.

2. The relocation of requirements will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transferred from the Technical Specifications to the Technical Specifications Bases are the same as the existing Technical Specifications. Since any future changes to these requirements in the Technical Specifications Bases will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

3. The proposed change modifies the surveillance frequency for drywell bypass leakage and associated air lock surveillances. Reliability of drywell integrity is evidenced by the measured leakage rate during past drywell bypass leakage surveillances. Appropriate design basis assumptions will be upheld, even when combined with the complementary bypass leakage surveillances as proposed. Drywell integrity will continue to be tested by means of the proposed periodic drywell bypass leakage test, performance of the drywell air lock door latching and interlock mechanism surveillance, and performance of additional surveillances including excercising of drywell isolation valves. The combination of these surveillances will provide adequate assurance that drywell bypass leakage will not exceed the design basis limit. Margins of safety would not be reduced unless leakage rates exceeded the design allowable drywell bypass leakage limit. Therefore, the proposed change does not cause 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: October 26, 1995

Description of amendment request: The proposed amendment would revise the technical specifications for sixteen editorial changes and would delete the requirement for a program to prevent and detect Asiatic Clams (Corbicula) in the service water system (SWS). The editorial changes covers such things as removing systems or components that do not exist in the River Bend Station, correcting typographical errors, correcting to be consistent with the writers guide for Improved Technical Specifications, adding descriptions for systems to make them clear, and wording changes to be consistent with approved facility operations. The Corbicula program is no longer needed because the facility has been modified and SWS no longer takes water from the Mississippi River; source of the larvae and infestation.

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:

EDITORIAL CHANGES

The purposed changes involves reformatting, renumbering and rewording of the existing Technical Specifications. The reformatting, renumbering and rewording process involves no technical changes to existing Technical Specifications. As such, these changes are administrative in nature and do not impact initiators of 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 evaluated.

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

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. These changes are administrative in nature. As such, no question of safety is involved, and the changes do not involve a significant reduction in a margin of safety.

CORBICULA PROGRAM

The proposed change deletes the program associated with the prevention and detection of Asiatic Clams (Corbicula) based upon improvements to the non-safety related Normal Service Water System (SWS). The source of makeup water to the SWS is no longer the Mississippi River, which is the source of Asiatic Clams. Demineralized water or well water is used eliminating the source of asiatic clams. To prevent biofouling SWS is treated with chlorine/bromine. This program is not considered as an initiator for any previously evaluated accident. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

The proposed change introduces no new mode of plant operation and it does not involve a physical modification to the plant. The possibility of the SES becoming contaminated by any other means is highly unlikely since it is a ''closed-loop'' system. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Prevention of Asiatic Clam infestation in the SWS and associated safety-related equipment is ensured by the ''closed-loop'' design of the SWS. Post Refuel Outage (RF–4) inspections of the safety-related heat exchangers that interface with the ''closed-loop'' SWS have shown no evidence of clam infestations. Therefore, the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: November 20, 1995

Description of amendment request: The proposed amendment would revise the technical specifications to eliminate the response time testing requirements for selected Reactor Protection System Instrumentation.

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 purpose of the proposed Technical Specification (TS) change is to eliminate response time testing requirements for selected components in the Reactor Protection System (RPS). The Boiling Water reactors Owners' Group (BWROG) has completed an evaluating which demonstrates that response time testing is redundant to the other TS-required testing. These other tests, in conjunction with actions taken in response to NRC Bulletin 90–01, ''Loss of Fill-Oil in Transmitters Manufactured by Rosemount,'' and Supplement 1, are sufficient to identify failure modes or degradation in instrument response times and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by the other TS-required testing. This evaluation was documented in NEDO–32291, ‘‘System Analyses for Elimination of Selected Response Time Testing Requirements,’’ January 1994. Entergy Operations, Inc. (EOI) has confirmed the applicability of this evaluation to River Bend Station (RBS). In addition EOI will complete the actions identified in the NRC staff's safety evaluation of NEDO–32291.

Because of the continued application of other existing TS-required tests such as channel calibration, channel checks, channel functional tests, and logic system functional tests, the response time of these systems will be maintained within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. The proposed changes do not affect the capability of the associated systems to perform their intended function within their required response time, nor do the proposed changes themselves affect the operation of any equipment. As a result, EOI has concluded that the proposed changes do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

The proposed changes only apply to the testing requirements for the components identified above and do not result in any physical change to these or other components or their operation. As a result, no new failure modes are introduced. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accidents previously evaluated.

The current TS-required response times are based on the maximum allowable values as assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed changes do not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for the plant safety analyses. The potential failure modes for the components within the scope of this request were evaluated for impact on instrument response time. This evaluation confirmed that, with the exception of loss of fill-oil of Rosemount transmitters, the remaining TS-required testing is sufficient to identify failure modes or degradation in instrument response times and ensure operation of the instrument within the scope of this request is within acceptable limits. The actions taken in response to NRC Bulletin 90–09 and Supplement 1 are adequate to identify loss of fill-oil failures of Rosemount transmitters. As a result, it has been concluded that plant and systems response to an initiating event will remain in compliance with the assumptions of the safety analysis.

Further, although not explicitly evaluated, the proposed changes will provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing the potential for safety system actuations, reducing plant shutdown risk, limiting radiation exposure to plant personnel, and eliminating the diversion of key personnel resources to conduct unnecessary testing. Therefore, EOI has concluded that this request will result in an overall increase 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

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

Date of amendment request: September 22, 1995

Description of amendment request: The proposed amendment would modify a requirement of the Seabrook Station, Unit No. 1 Technical Specifications. Specifically, the proposed amendment would change the ACTION referenced in Table 3.3–3, Engineered Safety Features Actuation System Instrumentation, for Functional Unit 8.b, Automatic Switchover to Containment Sump/RWST Level Low-Low. The ACTION requirement would be changed to ACTION 15 from ACTION 18. ACTION 15 requires an inoperable channel to be placed in bypass (with no time limit specified) while ACTION 18 requires an inoperable channel to be placed in the tripped condition within 6 hours.

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

A. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed change would result in an inoperable Functional Unit 8.b. protective channel being placed in the bypassed condition vice tripped condition. Functional Unit 8.b. is not involved in any accident initiation sequence; therefore, the probability of a previously-analyzed accident is not increased. Placing an inoperable Functional Unit 8.b. in bypass vice trip reduces the probability of premature opening of the containment building sump isolation valves thereby reducing the potential for increasing the consequences of a previously-analyzed accident. Thus, the consequences of a previously-analyzed accident is not increased.

B. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the change does not reduce the minimum required number of channels of instrumentation to be operable. The change does not alter the function of or affect the failure modes of Functional Unit 8.b. instrumentation channels. The proposed change does not otherwise affect the manner by which the facility is operated, and it does not involve any changes to equipment or features which affect the operational characteristics of the facility.

C. The change does not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the change does not reduce the minimum required number of channels of instrumentation to be operable, and it does not involve any changes to equipment or features which affect the operational characteristics of the facility. Therefore, the protection previously provided remains unchanged.

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

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

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141–0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 26, 1995, supplemented and revised October 20, 1995.

Description of amendment request: The proposed changes would modify TS 3.8.1.1., ‘‘Electrical Power Systems, A.C. Sources, Operating,’’ TS 3.8.1.2, ‘‘Electrical Power Systems, Shutdown,’’ TS 3.8.2.2, ‘‘Electrical Power Systems, A.C. Distribution - Shutdown,’’ and TS 3.8.2.4, ‘‘Electrical Power Systems, D.C. Distribution - Shutdown,’’ to provide operational flexibility as well as consistency between action statements and to eliminate certain surveillance requirements that are not applicable in Modes 5 or 6.

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:

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve an SHC. The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the change would not:

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

The proposed change to Surveillance Requirement 4.8.1.1.1 is being made because presently, the surveillance requirement for demonstrating offsite sources are operable states that ''two'' independent circuits are required. The surveillance requirement is referenced for both operating and shutdown modes. While it is accurate for operating modes, it is inconsistent with the limiting condition for operation for shutdown. The proposed change is safe because it renders the surveillance requirement consistent with the applicable limiting condition for operation (i.e., operating or shutdown) and eliminates a potential source of confusion.

The change to Surveillance Requirement 4.8.1.2 and Technical Specification 3.8.2.2 merely clarifies the diesel generator surveillance and operability requirements for Modes 5 and 6 and renders action statements for related technical specification sections consistent with and appropriate for operational Modes 5 and 6.

Regarding diesel generator surveillance requirements, automatic A.C. power for LNP events in Modes 5 and 6 is not required. This is validated by the fact that the undervoltage sensors are only required to be operable in Modes 1, 2 and 3 to meet technical specifications. Because the undervoltage sensors provide the logic that results in actuation of the sequencer, it follows that the sequencer need not be operable in Modes 5 and 6. Accordingly, the sequencer is not required to support operability of the available diesel generator in Modes 5 and 6. Further, because SIAS is blocked in Modes 5 and 6, automatic start of the diesel generator upon receipt of a SIAS is similarly not required to support operability of the diesel generator in Modes 5 and 6.

Additionally, operation of the diesel generator in parallel with the system during Modes 5 and 6 is not required to perform its intended safety function. In fact, such operation may compromise both sources as the result of a single event.

Since automatic A.C. power is not credited in the mitigation of Mode 5 and 6 events and accidents, such as fuel handling accidents, there is no increase in the probability or consequences of previously evaluated accidents.

The action statement in Technical Specification 3.8.2.2 has been revised to cite actions that are more appropriate for Modes 5 and 6 for Millstone Unit No. 2. This is due to the ability to maintain the plant in a safe condition without needing to automatically load the diesel generator through the sequencers in Modes 5 and 6. In addition, the proposed change is consistent with the CE Owner's Group Standard Technical Specification and with other Millstone Unit No. 2 action statements. Consequently, there is no increase in the probability or consequences of previously evaluated accidents.

The change to TS 3.8.2.4 merely renders the action statement consistent with, and appropriate for, operational Modes 5 and 6.

Since D.C. power is not credited in the mitigation of Mode 5 and 6 events and accidents, such as fuel handling accidents, there is no increase in the probability or consequences of previously evaluated accidents.

The action statement in TS 3.8.2.4 has been revised to cite actions that are more appropriate for Modes 5 and 6 for Millstone Unit No. 2. This is due to the ability to maintain the plant in a safe condition without D.C. power distribution available in Modes 5 and 6. In addition, the proposed change is consistent with the CE Owner's Group Standard Technical Specifications (NUREG–1432) and with other Millstone Unit No. 2 action statements. Consequently, there is no increase in the probability or consequences of previously evaluated accidents.

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

The proposed changes do not alter or affect the design, function, failure mode, or operation of the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes to the technical specifications provides greater consistency between the action statements and clarifies which surveillance requirements are required in Modes 5 and 6. Since the diesel generators are not required to be loaded automatically in Modes 5 and 6, and since it is part of our shutdown risk management program to assure that adequate cooling is able to be provided, and since the diesel will still be verified to start and achieve rated speed, the proposed changes to the technical specifications do not reduce the margin of safety.

The proposed change to the TS provides greater consistency among action statements during Modes 5 and 6. Since the D.C. distribution system is not credited in the mitigation of Mode 5 and 6 events and accidents, and since it is part of our shutdown risk management program to assure that adequate fuel cooling is able to be provided, the proposed change to the TS does not reduce the margin of safety.

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

Local Public Document Room location: 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: June 27, 1995, as supplemented July 21, 1995

Description of amendment request: The amendment revises the Technical Specifications (TS) to relocate TS requirements for the containment purge exhaust and supply valves, and to remove a duplicate testing requirement for the safety injection input from engineered safety features from 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:

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

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

The first proposed change relocates the operability and surveillance requirements for the containment high range radiation monitors from Technical Specification Section 3.3.3 to Technical Specification Section 3.3.2. The proposed changes are administrative in nature. The proposed changes do not alter the way any structure, system, or component functions and do not modify the manner in which the plant is operated and do not involve any physical changes to the plant.

The second proposed modification will delete the testing requirement for functional unit 16, ''Safety Injection Input from ESF,'' of Table 4.3–1 because the logic circuitry that processes the safety injection signals and produces a reactor trip is tested under functional unit 19 ''Automatic Trip and Interlock Logic,'' and the testing is performed on a more frequent basis (i.e., on a monthly staggered bases versus on an 18-month frequency). In addition, the same logic testing is accomplished with an 18-month TADOT of functional unit 1.a of Table 4.3–2 and with a monthly staggered actuation logic testing of functional unit 16 of Table 4.3–2. This testing ensures that operability of the logic under functional unit 16 of Table 4.3–1 is verified. The other tests will continue to verify the operability of the reactor trip system and that a reactor trip will be initiated when required.

Therefore, there is no change in the potential for an increase in the consequences of an accident previously analyzed.

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

The proposed changes do not affect the operation or response of any plant equipment or introduce any new failure mechanisms. The proposed elimination of the testing requirement line item does not affect the test results since the logic circuitry that processes the safety injection signal and produces a reactor trip will be tested and is tested under functional unit 19 of Table 4.3–1. As such, the changes do not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed changes do not have any adverse impact on the protective boundaries nor do they affect the consequences of any accident analyzed. The operability and surveillance requirements, although relocated to other technical specifications, will still ensure that the system (the radiation monitors) is tested and within limits. The proposed elimination of the testing equipment will not change the performance or operating conditions of the safety systems. The operable reactor trip system instrumentation ensures that the assumptions in the Bases of the Technical Specifications are not affected and ensures that the margin of safety is not reduced. Therefore, the proposed changes do not reduce the margin of safety.

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

Local Public Document Room location: 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

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

Date of amendment requests: November 14, 1994

Description of amendment requests: The proposed amendment would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, for the slave relay test frequency from quarterly (Q) to refueling (R). The request would also remove table notation 4 from Table 4.3–2. The associated Bases would also be appropriately revised.

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 results of WCAPs 14117 and 13878 demonstrate that slave relays are highly reliable. The WCAPs also provide guidance to assure that slave relays remain highly reliable. The aging assessment concludes that the age/temperature-related degradation of all ND relays, and NE relays produced after May 1990, is sufficiently slow such that a refueling frequency surveillance interval will not significantly increase the probability of slave relay failures. Finally, the evaluation of the interposing slave relays in the emergency diesel generator start circuitry, control room ventilation and auxiliary building ventilation realignments, steam generator blowdown isolation and radwaste isolation systems has concluded that based on the tests of the interposing relays performed during other equipment testing, reasonable assurance is provided that failures will be identified if the associated slave relays are tested on a refueling frequency.

The removal of table notation 4 from TS Table 4.3–2 is an administrative change that eliminates unnecessary redundancy from the TS and does not affect plant operation.

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

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

The proposed changes do not alter the performance of the ESFAS mitigation systems assumed in the plant safety analysis. Changing the interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios.

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

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

The proposed changes do not affect the total ESFAS response assumed in the safety analysis since the reliability of the slave relays will not be significantly affected by the increased surveillance frequency.

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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: August 18, 1995, as supplemented on November 1, 1995

Description of amendment request: The proposed amendment would revise the Operating License and Technical Specifications to allow for a power uprate to 2900 MWt. The current maximum power level is 2775 MWt.

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 or consequences of an accident previously evaluated is not significantly increased.

Implementation of uprate power operation does not contribute to any accident evaluated in the FSAR [Final Safety Analysis Report]. The NSSS [Nuclear Steam Supply System] Components (RV [reactor vessel], RCPs [reactor coolant pumps], CRDMs [control rod drive mechanisms], SGs [steam generators], and piping) are compatible with the revised operating conditions. These components have been reanalyzed and the results show that ASME [American Society of Mechanical Engineers] Code requirements remain satisfied and are within the current Licensing Basis.

Interfacing Systems which are important to safety are not adversely impacted and will continue to perform their design function. Overall secondary plant performance is not significantly altered by the proposed changes.

The revision to the Pressure Temperature Limits will not adversely impact the RCS [reactor coolant system] Pressure Boundary. The length of time these curves will be applicable, due to increased neutron fluence, is being reduced. Before the 13 Effective Full Power Years have elapsed, new curves will be generated to reflect the analysis of the specimen capsule and will be derived utilizing NRC approved methodology.

Therefore, since the Reactor Coolant pressure boundary integrity and system functions are not adversely impacted, the probability of occurrence of an accident evaluated in the VCSNS [Virgil C. Summer Nuclear Station] FSAR will be no greater than the original design basis of the plant.

An extensive analysis has been performed to evaluate the consequences of the following accident types currently evaluated in the VCSNS FSAR:

- Non-LOCA [loss-of-coolant accident] Events
- Large Break and Small Break LOCA
- Steam Generator Tube Rupture

With the [delta]75 SGs and revised operating conditions, the calculated results (i.e., DNBR [departure from nucleate boiling ratio], Primary and Secondary System Pressure, Peak Clad Temperature, Metal Water Reaction, Challenge to Long Term Cooling, Environmental Conditions Inside and Outside containment, etc.) for the accidents are similar to those currently reported in the VCSNS FSAR and remain within applicable Regulatory Acceptance Criteria. Select results (i.e., Containment Pressure during a Steam Line Break, Minimum DNBR for Rod Withdrawal from Subcritical, etc.) are slightly more limiting than those currently reported in the FSAR due to the use of the assumed operating conditions with the [delta]75 SGs and in some cases, use of an uprated core power of 2900 MWt. However, in all cases, the calculated results do not challenge the integrity of the primary/secondary/containment pressure boundary and remain within the regulatory acceptance criteria applied to VCSNS's current licensing basis.

Given that calculated radiological consequences are not significantly higher than current FSAR results and remain well within 10 CFR 100 limits, it is concluded that the consequences of an accident previously evaluated in the FSAR are not significantly increased.

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

Uprate power operation will not introduce any new accident initiator mechanisms. Structural integrity of the RCS is maintained during all plant conditions through compliance with the ASME code and 10 CFR 50 Appendix G requirements. Design requirements of auxiliary systems are met with the RSGs [replacement steam generators] and uprate power operation. No new failure modes or limiting single failures have been identified. Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

Although uprate power operation will require changes to the VCSNS Technical Specifications, the proposed changes are supported by extensive LOCA, NON-LOCA and SGTR [steam generator tube rupture] analyses. These analyses show acceptable consequences with margin to the applicable regulatory limits. All equipment required to function during accident conditions has been shown to remain qualified and thus will perform their design function, and all components remain in compliance with the codes and standards in effect when VCSNS was originally licensed (with the exception of the replacement steam generators which use the 1986 ASME Code Section III Edition).

Low Temperature Overpressure transients which could challenge RCS structural integrity are not impacted by the revision to the Pressure Temperature Limitations Curves. The curves are not directly impacted, the changes do not reduce any margin of safety.

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

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

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: August 29, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications for allowable values and trip setpoints for selected plant process instrumentation. The new allowable values/setpoints are in accordance with the instrument setpoint methodology accepted by the NRC staff in a letter dated July 18, 1995.

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 revised Trip Setpoints and Allowable Values are more conservative than those currently approved in the Technical Specifications. Therefore, any proposed system or component actuations will occur earlier, resulting in a more conservative plant response. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the Technical Specifications does not introduce any new components nor does it modify the design of any existing components. Other than making Trip Setpoints and Allowable Values of existing instrumentation more conservative, the change does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. Thus, the possibility of a new or different kind of accident previously evaluated is not created.

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

Since the proposed revised Trip Setpoints and Allowable Values are more conservative than the existing values, the margin of safety would be increased by issuance of the changes. Thus, the proposed change does not result in 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: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: November 2, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications to allow 120 volt AC buses EV–1–A and EV–1–B to be energized from either their normal inverter power supply or from their alternate power supply.

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:

These buses are not used as the initiator of any analyzed accidents. Therefore, the probability of any previously evaluated accident has not increased. If an accident were to occur while the buses are supplied from the alternate power supply, there would be no change in the analyzed accident scenario since even in the event of a loss of offsite power event, the safety functions would be completed. Thus, the consequences of any previously evaluated accident have not increased.

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

The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it 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:

This change does not involve a significant reduction in a margin of safety since the proposed change maintains a safety related, diesel-backed power supply to these buses whether the power is supplied from the inverters or from the alternate power supply. If a loss of offsite power event were to occur while the buses were supplied from the alternate power source, the safety functions being performed by components supplied from these buses would occur. Thus, there has been no reduction in the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: November 2, 1995

Description of amendment request: The proposed amendment to the Perry Nuclear Power Plant Technical Specifications revises those specifications associated with handling irradiated fuel in Primary Containment and the Fuel Handling Building, and selected specifications associated with CORE ALTERATIONS. Specifically, analysis identifies that only —recently— irradiated fuel contains sufficient fission products to require OPERABILITY of accident mitigation features to meet the accident analysis assumptions. Analyses also show that accident mitigation features such as building INTEGRITY and engineered safety feature (ESF) ventilation systems are not required for CORE ALTERATION events.

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 requirements are imposed during specific activities which can be postulated to result in significant radioactive releases. The proposed APPLICABILITY requirements are consistent with either the original design basis analyses or with revised analyses performed to support this proposed amendment. Because the equipment controlled by the revised Specifications is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident.

Consistent with the original design basis analysis, the reanalysis concludes that radiological consequences of the fuel handling accident are well within the 10 CFR 100.11 limits, as defined by acceptance criteria in Standard Review Plan Section 15.7.4. The reanalysis has previously been submitted to the Nuclear Regulatory Commission for review, and NRC confirmatory calculations reached consistent results (reference NRC Safety Evaluation for License Amendment No. 35). The results of the CORE ALTERATION events other than the fuel handling accident remain unchanged from the original design basis, which showed that these events do not result in fuel cladding integrity damage or radioactive releases. Therefore, the proposed changes do not significantly increase the consequences of any previously evaluated accident.

Based on the above, the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

The proposed requirements are imposed when specific activities represent situations where significant radioactive releases can be postulated. The proposed APPLICABILITY requirements are consistent with design basis analyses. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accidident from any previously evaluated.

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

The proposed change imposes controls to ensure that during performance of activities which represent situations where radioactive releases are postulated, the radiological consequences are at or below the established licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Substantial conservatism is retained to ensure that the analysis adequately bounds all postulated event scenarios. The current margin of safety is retained.

Specifically, the margin of safety for the fuel handling accident is the difference between the 10 CFR 100 limits and the licensing limit defined by the Standard Review Plan (NUREG 0800), Section 15.7.4. The licensing limit is defined by the Standard Review Plan as being —well within— the 10 CFR 100 limits, with ''well within'' defined as 25% of the 10 CFR 100 limits for the fuel handling accident. Excess margin is the difference between the postulated doses and the corresponding licensing limit. In the NRCs initial licensing review of the Perry Nuclear Power Plant (NUREG–0887, Section 15.3.3), the NRC accepted the design and analyses based on the results of the analyses being well within the guideline values of 10 CFR 100.

The proposed APPLICABILITY requirements continue to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries as well as control room doses are at or below the corresponding licensing limit. The margin of safety is unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The margin of safety for the CORE ALTERATION events other than the fuel handling accident discussed above also remains the same as in the original design basis analyses, since the proposed changes do not impact on the Technical Specification requirements for systems needed to prevent or mitigate such CORE ALTERATION events.

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

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

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

NRC Project Director: Gail H. Marcus

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: October 6, 1995, and supplemented November 20, 1995

Brief description of amendments: The amendments revise the Technical Specifications by incorporating a new acceptance criterion for steam generator tubes with degradation in the tubesheet roll expansion region.

Date of issuance: November 21, 1995

Effective date: November 21, 1995

Amendment Nos.: 172 and 159

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 16, 1995 (60 FR 53648) The supplemental letter provided clarifying information that did not affect the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 21, 1995.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: August 10, 1995

Brief description of amendment: The amendment revises the Haddam Neck Technical Specification Section 3/4.4.3, ''Pressurizer,'' to add a footnote to allow the pressurizer level to be controlled, outside of the programmed level, between 25 to 50 percent, plus or minus 5 percent in Mode 3 when the reactor coolant system is borated to the required Mode 5 concentrations.

Date of Issuance: November 14, 1995

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

Amendment No.: 186

Facility Operating License No. DPR–61. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52928) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 14, 1995.

No significant hazards consideration comments received: No

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

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: September 13, 1995, as supplemented October 16,1995

Brief description of amendments: These amendments revise the Administrative Controls section of the BVPS–1 and BVPS–2 TSs to make them consistent with the requirements of the Offsite Dose Calculation Manual (ODCM). The ODCM was recently updated to reflect the radioactive liquid and gaseous effluent release limits and the liquid holdup tank activity limit of BVPS–1 License Amendment No. 188 and BVPS–2 License Amendment No. 70 which were issued June 12, 1995.

Date of issuance: November 21, 1995

Effective date: As of the date of issuance, to be implemented within 10 days.

Amendment Nos.: 194 and 77

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

Date of initial notice in Federal Register: September 22, 1995 (60 FR 49292) The October 16, 1995, letter did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the September 22, 1995, Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 21, 1995.

No significant hazards consideration comments received: No

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

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

Date of amendment request: February 14, 1994, as supplemented by letters dated July 25, August 15, and August 29, 1995

Brief description of amendment: The amendment changes the Appendix A Technical Specifications (TSs) to make them consistent with the revised 10 CFR Part 20, Standards for Protection Against Radiation.

Date of issuance: November 17, 1995

Effective date: November 17, 1995

Amendment No.: 116

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

Date of initial notice in Federal Register: March 30, 1994 (59 FR 14888) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 1995. The July 25, August 15, and August 29, 1995 letters provided clarifying information that did not change the initial propose no significance hazards consideration determination.

No significant hazards consideration comments received: No

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

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: May 12, 1995, as supplemented by letters dated July 6 and October 2, 1995.

Brief description of amendments: The amendments revise Technical Specification Surveillance Requirement 4.6.1.2 to add the provision that 10 CFR Part 50, Appendix J, applies, except as modified by NRC-approved exemptions.

Date of issuance: November 17, 1995

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

Amendment Nos.: 91 and 69

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

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35078) The July 6 and October 2, 1995, letters provided clarifying information that did not change the scope of the May 12, 1995, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 1995.

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, Docket No. 50–245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: July 28, 1995, as supplemented September 12, October 18, and October 31, 1995.

Brief description of amendment: In order to support a full-core offload as a normal end-of-cycle event, the amendment adds License Condition 2.C(6) and will require that: (1) the reactor be subcritical for at least 100 hours prior to the start of reactor refueling operations, (2) the spent fuel pool bulk temperature be maintained less than or equal to 140—F, and (3) two trains of shutdown cooling be operable during reactor refueling operations.

Date of issuance: November 9, 1995

Effective date: As of the date of issuance.

Amendment No.: 89

Facility Operating License No. DPR–21. Amendment revised the license.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45180) The September 12, October 18, and October 31, 1995, submittals provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment and Final No Significant Hazards Consideration Determination are contained in a Safety Evaluation dated November 9, 1995.

No significant hazards consideration comments received: No public comments received. A request for a hearing was received from We the People, the Seacoast Anti-Pollution League, the New England Coalition on Nuclear Pollution, and Donald Del Core of Uncasville, Connecticut.

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

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

Date of application for amendment: October 6, 1995, supplemented October 23, November 2, and November 15, 1995.

Brief description of amendment: The amendment adds footnotes to Action Statement (AS) 3.8.1.1.a of the Technical Specification (TS) and its bases to allow a one-time extension of the allowed outage time (AOT) for an inoperable offsite power source from the current 72 hours to 7 days.

Date of issuance: November 22, 1995

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

Amendment No.: 192

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

Date of initial notice in Federal Register: October 17, 1995 (60 FR 53812). The October 23, November 2, and November 15, 1995, letters provided clarifying information and slight modifications to the original request that were not outside the scope of the original notice and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1995.

No significant hazards consideration comments received: No

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

Northern States Power Company, Docket No. 50–282, Prairie Island Nuclear Generating Plant, Unit No. 1, Goodhue County, Minnesota

Date of application for amendment: January 10, 1995, as supplemented August 9 and September 20, 1995.

Brief description of amendment: The amendments revise the Prairie Island event monitoring instrumentation Technical Specifications and associated Bases to conform to Standard Technical Specifications for post-accident monitoring.

Date of issuance: November 9, 1995

Effective date: November 9, 1995, with full implementation within 30 days.

Amendment Nos.: 121/114

Facility Operating License No. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8753) The August 9 and September 20, 1995, letters provided updated Technical Specification pages and clarifying information in response to discussions with the staff during various teleconferences conducted during the review process. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 9, 1995.

No Significant hazards consideration comments received: No

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

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: December 2, 1994, as supplemented May 12, 1995.

Brief description of amendments: These amendments relocate the fire protection requirements from the Technical Specifications to the Updated Final Safety Analysis Report in accordance with the guidance in Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements,'' and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications.''

Date of issuance: November 20, 1995

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

Amendment Nos.: 104 and 68

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

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20524) The supplemental letter provided clarifying information and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 20, 1995.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: September 14, 1995 and supplemented by letter dated October 27, 1995

Brief description of amendments: These amendments revise the technical specifications by deleting Reactor Enclosure and Refueling Area Secondary Containment Isolation Valve Tables 3.6.5.2.1–1 and 3.6.5.2.2–1, and references to them, in accordance with Generic Letter 91–08, ''Removal of Component lists from Technical Specifications.'' The TS have been modified to state requirements in general terms that include the components listed in the tables removed from the TS.

Date of issuance: November 20, 1995

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

Amendment Nos.: November 20, 1995

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

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52934) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 20, 1995.

No significant hazards consideration comments received: No

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

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of application for amendments: October 4, 1995 (TS 368)

Brief description of amendment: The amendment delete requirements for daily checks for certain instruments that do not have indications, and provides editorial changes.

Date of issuance: November 13, 1995

Effective Date: November 13, 1995

Amendment No.: 202

Facility Operating License No. DPR–68: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52935) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 1995.

No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

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: August 7, 1995 (TS 95–03)

Brief description of amendments: The amendments address operation with a rod urgent failure condition, including limited operation with one control or shutdown bank inserted up to 18 steps below its insertion point. In addition, the surveillance interval for rod movement verifications has been increased from 31 to 92 days.

Date of issuance: November 21, 1995

Effective date: November 21, 1995

Amendment Nos.: 215 and 205

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45186) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 1995.

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: April 28, 1995

Brief description of amendment: The amendment removes the license conditions for the Transamerica Delaval, Inc. emergency diesel generators specified by paragraph 2.C.(9) and defined in Attachment 2 to the Operating License.

Date of issuance: November 16, 1995

Effective date: November 16, 1995

Amendment No.: 74

Facility Operating License No. NPF–58: This amendment revises the license.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29889) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 1995.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 23, 1995, and facsimile transmission dated October 31, 1995

Brief description of amendment: This amendment relocates TS 3/4.3.3.3, ''Seismic Instrumentation;'' TS 3/4.3.3.4, ''Meteorological Instrumentation;'' and TS 3/4.4.11, ''Reactor Coolant System Vents;'' and the Bases for each of the three sections from the TS to the Updated Safety Analysis Report, and eliminates the special reporting requirements for inoperable seismic and meteorological monitoring instrumentation from TS 6.9.2.

Date of issuance: November 14, 1995

Effective date: November 14, 1995, and shall be implemented not later than 90 days after issuance.

Amendment No.: 201

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

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39455) The October 31, 1995, facsimile transmission was clarifying in nature and did not affect the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 1995.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 7, 1995

Brief description of amendment: This amendment revises Technical Specification 3/4.9.4, Refueling Operations - Containment Penetrations;

Bases 3/4.9.4, Containment Penetrations; and Limiting Condition for Operation (LCO) 3.9.4.b to allow both doors of the containment personnel airlock to be open during core alterations or movement of irradiated fuel within the containment, provided that certain specified conditions are meet. Additional changes revise or clarify TS LCO 3.9.4.c, TS Action 3.9.4.a, and TS Surveillance Requirement 4.9.4, and modify the associated Bases.

Date of issuance: November 17, 1995

Effective date: November 17, 1995

Amendment No.: 202

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

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39454) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 1995.

No significant hazards consideration comments received: No

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

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

Date of amendment request: December 6, 1994

Brief description of amendments: These changes revise Technical Specifications to allow appropriate remedial action for high particulate levels in the diesel generator fuel oil inventory and other out-of-limit properties in new diesel generator fuel oil that has been added to the existing diesel generator fuel oil storage inventory.

Date of issuance: November 17, 1995

Effective date: November 17, 1995

Amendment Nos.: Unit 1 - Amendment No. 43; Unit 2 - Amendment No. 29

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 6311) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 17, 1995.

No significant hazards consideration comments received: No

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

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

Date of amendment request: January 13, 1995

Brief description of amendment: The amendment revises Technical Specifications (TS) 3.3.1 and 3.3.2 to relocate Tables 3.3–2 and 3.3–5, which provide the response time limits for the reactor trip system and the engineered safety features actuation system instruments, from the TS to the updated Final Safety Analysis Report (FSAR). The amendment also relocates the Bases discussion for TS 3.3.1 and TS 3.3.2 to Section 16.3 of the updated FSAR.

Date of issuance: November 22, 1995

Effective date: November 22, 1995, to be implemented within 30 days of issuance.

Amendment No.: 104

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

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8741) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1995.

No significant hazards consideration comments received: No

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

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: June 6, 1995

Brief description of amendment: The amendment modifies the Index of the WNP–2 Technical Specifications by deleting reference to the Bases pages.

Date of issuance: November 24, 1995

Effective date: November 24, 1995

Amendment No.: 143

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37102) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 24, 1995.

No significant hazards consideration comments received: No

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352

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

Date of application for amendments: September 13, 1995, and October 19, 1995, as supplemented by letter dated October 25, 1995

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.1, ''Definitions,'' TS Section 15.3.1.G, ''Operational Limitations'' (and basis), and TS Figure 15.2.1–2, ''Reactor Core Safety Limits, Point Beach Unit 2.'' The changes reduce the reactor coolant system raw measured total flow rate limit and reflect new reactor core safety limits for Unit 2.

Date of issuance: November 17, 1995

Effective date: November 17, 1995

Amendment Nos.: 165 and 169

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 54527 dated October 24, 1995). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by November 24, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated November 17, 1995.

No significant hazards consideration comments received: No

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

Date of amendment request: September 14, 1995

Brief description of amendment: The amendment revised Technical Specification 3/4.5.5 to increase the allowed outage time for adjustment of boron concentration for the refueling water storage tank from 1 hour to 8 hours.

Date of issuance: November 13, 1995

Effective date: November 13, 1995, to be implemented within 30 days of issuance.

Amendment No.: 91

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52936) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 1995.

No significant hazards consideration comments received: No

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

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 January 5, 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).

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

Date of application for amendments: November 9, 1995, as supplemented by letters dated November 13, 1995, and November 16, 1995

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.4.2, ''In-Service Inspection of Safety Class Components,'' to incorporate a new steam generator tube acceptance criterion for the Unit 2 steam generators. This criterion allows tubes that are degraded or defective in a location (within the tubesheet) that does not affect the structural integrity of the tube to remain in service. The applicable basis is also changed.

Date of issuance: November 22, 1995

Effective date: November 22, 1995

Amendment Nos.: 166 and 170

Facility Operating License Nos. DPR–24 and DPR–27. 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 November 22, 1995.

No significant hazards consideration comments received: No

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: Ernest L. Blake, Jr., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: Gail H. Marcus

Dated at Rockville, Maryland, this 29th day of November 1995.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 95–29540 Filed 12–5–95; 8:45 am]

BILLING CODE 7590–01–F

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–21547; No. 812–9652]

Southland Life Insurance Company, et al.

November 29, 1995.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of application for an order pursuant to the Investment Company Act of 1940 (the ''1940 Act'').

APPLICANTS: Southland Life Insurance Company (''Southland''), Southland Separate Account A1 (the ''Account''), and ING America Equities, Inc. (''ING Equities'').

RELEVANT 1940 ACT SECTIONS: Order requested pursuant to Section 6(c) of the 1940 Act granting exemptions from the provisions of Sections 26(a)(2)(C) and 27(c)(2) thereof.

SUMMARY OF APPLICATION: Applicants seek an order permitting the deduction of mortality and expense risk and enhanced death benefit charges from the assets of: (a) The Account in connection with the offer and sale of certain variable annuity contracts (''Existing Contracts''); (b) the Account in connection with the issuance of variable annuity contracts that are substantially similar in all material respects to the Existing Contracts (''Future Contracts,'' together with Existing Contracts, the ''Contracts''); and (c) any other separate account established in the future by Southland in connection with the issuance of Contracts (''Future Account'').

FILING DATE: The application was filed on June 29, 1995. Applicants have undertaken to amend the application during the notice period to make the representations contained herein.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Secretary of the Commission and serving Applicants with a copy of the request, personally or by mail. Hearing requests must be received by the Commission by 5:30 p.m. on December 26, 1995, and must be