(Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the resolution of the MSRP issues.

*1:30 P.M.–3:00 P.M.: Meeting with the Director of the Office of Nuclear Reactor Regulation (NRR)* (Open)—The Committee will hear presentations by and hold discussions with Mr. William Russell, NRR Director, on items of mutual interest, including the following: Risk/Performance-Based Regulations, Risk-Based Inspection Program, Activities of the Nuclear Industry in Support of the Risk/ Performance-Based Regulations, AP600 and SBWR review status, and ASME piping code review.

*3:00 P.M.–3:30 P.M.: Report of the Planning and Procedures Subcommittee* (Open/Closed)—The Committee will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS staff members.

A portion of this session may be closed to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of this Advisory Committee, and matters the release of which would constitute a clearly unwarranted invasion of personal privacy.

*3:45 P.M.–4:15 P.M.: Future ACRS Activities* (Open)—The Committee will discuss recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings.

*4:15 P.M.–4:30 P.M.: Reconciliation of ACRS Comments and Recommendations* (Open)—The Committee will discuss responses of the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports. These responses are expected to be received from the EDO before the meeting.

*4:30 P.M.–6:45 P.M.: Preparation of ACRS Reports* (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting as well as a proposed ACRS report on resolution of Generic Safety Issue-78, ‘‘Monitoring of Fatigue Transient Limits for the Reactor Coolant System’’.

*Friday, December 8, 1995*

*8:30 A.M.–8:35 A.M.: Opening Remarks by the ACRS Chairman* (Open)—The ACRS Chairman will make opening remarks regarding conduct of the meeting.

*8:35 A.M.–9:15 A.M.: Preparation of ACRS Reports* (Open)—The Committee will continue its discussion of proposed ACRS reports on matters considered during this meeting.

*9:30 A.M.–10:00 A.M.: Preparation for Meeting with the NRC Chairman* (Open)—The Committee will select items that may be discussed with the NRC Chairman.

*10:00 A.M.–11:00 A.M.: Meeting with the NRC Chairman* (Open)—The Chairman will meet with the Committee to discuss her regulatory agenda and philosophy, and other items of mutual interest.

*11:15 A.M.–12:15 P.M.: Preparation for Meeting with the NRC Commissioners* (Open)—The Committee will prepare for meeting with the NRC Commissioners to discuss items of mutual interest including, Rulemaking to amend 10 CFR 50.48, Fire Protection, Nondestructive Examination Techniques, and National Academy of Sciences/National Research Council Study on Digital Instrumentation and Control.

*1:30 P.M.–3:00 P.M.: Meeting with the NRC Commissioners* (Open)—The Committee will meet with the NRC Commissioners in the Commissioners' Conference Room, One White Flint North, to discuss items of mutual interest including those noted above.

*3:15 P.M.—3:45 P.M.: Election of Officers for Calendar Year 1996* (Open)—The Committee will elect Chairman and Vice Chairman to the ACRS, and Member-at-Large to the Planning and Procedures Subcommittee for Calendar Year 1996.

*3:45 P.M.–5:30 P.M.: Preparation of ACRS Reports* (Open)—The Committee will continue its discussion of proposed ACRS reports on matters considered during this meeting.

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on September 27, 1995 (60 FR 49925). In accordance with these procedures, oral or written statements may be presented by members of the public, electronic recordings will be permitted only during the open portions of the meeting, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. Sam Duraiswamy, Chief, Nuclear Reactors Branch, at least five days before the meeting, if possible, 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 Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief of the Nuclear Reactors Branch prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Chief of the Nuclear Reactors Branch if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) P.L. 92–463, I have determined that it is necessary to close portions of this meeting noted above to discuss matters that relate solely to the internal personnel rules and practices of this Advisory Committee per 5 U.S.C. 552b(c)(2), and to discuss matters the release of which would constitute a clearly unwarranted invasion of personal privacy per 5 U.S.C. 552b(c)(6).

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. Sam Duraiswamy, Chief, Nuclear Reactors Branch (telephone 301/415–7364), between 7:30 A.M. and 4:15 P.M. EST.

ACRS 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 ACRS meeting dates for Calendar Year 1996 are provided below:

| ACRS meeting No. | 1996 ACRS meeting dates |
|---|---|
| 428 | February 8–10, 1996 |
| 429 | March 7–9, 1996 |
| 430 | April 11–13, 1996 |
| 431 | May 23–25, 1996 |
| 432 | June 20–22, 1996 |
| 433 | August 8–10, 1996 |
| 434 | September 12–14, 1996 |
| 435 | October 10–12, 1996 |
| 436 | November 7–9, 1996 |
| 437 | December 5–7, 1996 |

Dated: November 20, 1995.

Andrew L. Bates

*Advisory Committee Management Officer*

[FR Doc. 95–28836 Filed 11–24–95; 8:45 am]

BILLING CODE 7590–01–P

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 28, 1995, through November 9, 1995. The last biweekly notice was published on Wednesday, November 8, 1995 (60 FR 56361).

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 December 27, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

*Date of amendment request:* October 20, 1995.

*Description of amendment request:* The proposed one-time amendment would revise the Calvert Cliffs Nuclear Power Plant, Unit No. 1, (CC–1) Technical Specifications (TSs) by extending certain 18-month instrument surveillance intervals by a maximum of 39 days to March 31, 1996. The instruments involved are included in the reactor protective system, engineered safety features actuation system, power-operated relief valves, low-temperature overpressure protection system, remote shutdown instruments, post-accident monitoring, radiation monitoring, and containment sump level instruments.

The Commission issued Amendment No. 208 to Facility Operating License No. DRP–53 and Amendment No. 186 to Facility Operating License No. DRP–69 for the CC–1/2, respectively. The amendments permanently extended the surveillance intervals for the instruments described above from 18 months to 24 months after a specified number of the instruments had been replaced. The amendments were effective immediately and to be implemented on CC–2 within 30 days, but not implemented on CC–1 until its restart after the spring 1996 refueling outage. All of the instruments identified for replacement on CC–2 have been replaced, but those identified for replacement on CC–1 have not been replaced, thus, the reason for the later implementation date. The proposed one-time amendment is needed prior to Amendment No. 208 being implemented because of a change in the refueling schedule. The licensee has provided technical justification to allow operation for an additional short-time period of up to a maximum of 39 days.

CC–1 was initially scheduled to begin its refueling outage on February 16, 1996, which would have been within the time frame necessary to perform the required 18-month instrument surveillances currently required for the instruments identified above. The licensee has recently rescheduled the refueling outage for CC–1 to start March 15, 1996, several months after the initial amendment request and after consultation with the Pennsylvania-New Jersey-Maryland power pool. The revised schedule will allow the maximum use of the available fuel in the CC–1 reactor core and will also allow the unit to operate for an additional period of about 1 month during a period of potentially high power demand. In addition, the delay will allow more time to plan and prepare for the upcoming refueling outage. Performing the required instrument surveillances at power would present an unwarranted personnel safety risk and, in some cases, the surveillances cannot be done during power operation because they would cause a unit trip. This proposed one-time amendment will be superseded by Amendment No. 208 when it is implemented.

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

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

The proposed one-time change would extend 18-month instrument surveillance intervals by a maximum of 39 days to March 31, 1996, for specific Reactor Protective System (RPS), Engineered Safety Features Actuation System (ESFAS), Power-Operated Relief Valve, Low Temperature Overpressure Protection (LTOP), Remote Shutdown, Post-Accident Monitoring (PAM), Radiation Monitoring, and Containment Sump Level instruments.

The purpose of the RPS is to effect a rapid reactor shutdown if any one or a combination of conditions deviates from a pre-selected operating range. The system functions to protect the core and the Reactor Coolant System (RCS) pressure boundary. The purpose of the ESFAS is to actuate equipment which protects the public and plant personnel from the accidental release of radioactive fission products if an accident occurs, including a loss-of-coolant accident, main steam line break, or loss of feedwater event. The safety features function to localize, control, mitigate, and terminate such incidents in order to minimize radiation exposure to the general public. The PAM instruments provide the Control Room operators with primary information necessary to take manual actions, as necessary, in response to design basis events, and to verify proper system response to plant conditions and operator actions. The purpose of the Remote Shutdown System is to provide plant parameter indications to operators on a Remote Shutdown Panel to be used while placing and maintaining the plant in a safe shutdown condition in the event the Control Room is uninhabitable. The indications are used to verify proper system response to plant conditions and operator actions. The LTOP System protects against RCS overpressurization at low temperatures by a combination of administrative controls and hardware. Power-Operated Relief Valves are set to lift before pressurizer safety valves, and subsequently reseat to minimize the release of reactor coolant from the RCS. The Containment Sump High Level Alarm System provides an alarm in the Control Room to provide one of the available indications of excessive RCS leakage during normal plant operation. The Containment Area High Range Radiation Monitoring System provides an indication of high radiation levels in containment.

Failure of any of these systems is not an initiator for any previously evaluated accident. Therefore, the proposed change would not involve an increase in the probability of an accident previously evaluated.

Surveillance and maintenance history has demonstrated good capability for identifying adverse operation by individual instruments. Baltimore Gas and Electric Company has the capability to respond to an inoperable instrument by following the Technical Specification Actions for an inoperable instrument or by performing a channel calibration with the Unit at full power. However, calibration of all the instruments at power is not desirable because of personnel safety, personnel radiation protection goals, and plant reliability concerns.

These factors provide assurance that the requested surveillance extension will not adversely affect our ability to detect degradation of the instruments. Also, either analysis is available to show the instruments will operate properly during the requested surveillance extension, or the surveillance program has shown that problems will be identified and addressed appropriately. Therefore, these channels will be able to perform the functions assumed in the safety analysis, and there is no significant increase in the consequences of an accident previously evaluated.

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

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

This requested increase in surveillance interval for RPS, ESFAS, Power-Operated Relief Valve, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instrument surveillances does not involve a significant change in the design or operation of the plant. No plant hardware is being modified as part of the proposed change. The proposed change also does not involve any new or unusual actions by plant operators. Therefore, this change would not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

The RPS, ESFAS, Power-Operated Relief Valve, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instruments are designed to provide actuation signals and/or indications to ensure appropriate action is taken in response to design basis accidents. Channel checks, channel functional tests and routine comparison of the redundant and independent parameter indications provides a reliable indication of instrument operation. Also, either analysis is available to show the instruments will operate properly during the requested surveillance extension, or instrument surveillance program has shown that problems will be identified and addressed appropriately. During the requested extension, these systems will be available to perform the functions assumed in the Safety Analysis. Surveillance and maintenance history have demonstrated good capability for identifying adverse operation by individual instruments. Baltimore Gas and Electric Company has the capability to respond to such adverse operation, including performing channel calibrations at power. However, such work on all the instruments is not desirable because of personnel safety, personnel radiation protection goals, and plant reliability concerns. Extending the surveillance interval provides additional possibility for instrument components to malfunction by means such as drift or instrument failure, which could allow plant parameters to exceed design bases assumptions. We have determined that the effect of the surveillance interval extension on safety is small, and operation of the instruments in the extended interval would not invalidate any assumption in the plant licensing basis.

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

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

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

*NRC Project Director:* Ledyard B. Marsh.

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

*Date of amendment request:* October 3, 1995.

*Description of amendment request:* The proposed amendments would revise the Technical Specifications (TSs) for both stations to implement 10 of the line item TS improvements recommended in Generic Letter (GL) 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993. The proposed changes also include editorial changes on the affected TS pages.

The proposed changes from GL 93–05 are the following: (1) TS 4.1.3.1.2 (GL 93–05, Item 4.2), extending the interval for checking the operability of each full-length rod not fully inserted in the core from 31 days to 92 days; (2) Table 4.3–3 (GL 93–05, Item 5.14), extending the interval for the digital channel operational test for radiation monitoring instrumentation in the table from monthly to quarterly; (3) TS 4.4.3.2 (GL 93–05, Item 6.6), extending the interval between current tests of the required groups of pressurizer heaters from 92 days to each refueling outage; (4) TS 4.4.6.2.2.b (GL 93–05, Item 6.1), extending the time the plant may be in cold shutdown before pressure isolation valve testing is required, prior to entry into Operational Mode 2, from 72 hours to 7 days; (5) TS 4.5.1.1.b (GL 93–05, Item 7.1), revising the requirement to verify the boron concentration in an accumulator within 6 hours of any volume increase to the accumulator (greater than or equal to 70 gallons) so that the verification is not required when the volume increase is from the refueling water storage tank (RWST) and the RWST has not been diluted since verifying that the boron concentration of the RWST is within the concentration limits for the accumulators; (6) TS 4.6.2.1 (GL 93–05, Item 8.1), extending the interval between tests to verify each containment spray nozzle is unobstructed from 5 years to 10 years; (7) TS 4.6.4.1 (GL 93–05, Item 5.4), extending the interval for testing each hydrogen monitor for combustible gas control from 31 days to 92 days for the analog channel operational test, and from 92 days to each refueling outage for channel calibration; (8) TS 4.6.4.2 (GL 93–05, Item 8.5), extending the interval between tests to demonstrate operability of the hydrogen recombiner system from 6 months to once each refueling outage; (9) TS 4.7.1.2.1.a (GL 93–05, Item 9.1), extending the interval between tests of the auxiliary feedwater pumps from 31 days to 92 days on a staggered test basis; and (10) TS 4.11.2.6 (GL 93–05, Item 13), extending the interval for determining the quantity of radioactivity contained in each gas decay tank, when radioactivity is being added to the tanks, from 24 hours to 7 days, with the 24-hour frequency maintained during the primary coolant degassing operation. The editorial changes are the following: (1) TS 4.4.6.2.1.c, changes the word ''from'' to the word ''to,'' (2) TS 4.5.1.1.c, the change clarifies that the motor control center compartment is for each accumulator isolation valve, (3) TS 4.5.1.2, deletes the footnote because the operating cycle in the footnote is over for each unit, and (4) TS 4.7.1.2.1.a.2 and 4.7.1.2.1.c, renumbers and rephrases (only TS 4.7.1.2.1.a.2) other surveillance requirements for the auxiliary feedwater pumps because of the proposed change to TS 4.7.1.2.1.a to implement GL 93–05, Item 9.1.

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

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

The changes are consistent with GL 93–05 and NUREG–1366 [''Improvements to Technical Specifications Surveillance Requirements,'' December 1992. In GL 93–05, the staff stated that it concluded, in performing the study documented in NUREG–1366, that safety can be improved, equipment degradation decreased, and an unnecessary burden on licensee personnel eliminated by reducing the frequency of certain testing required in the Technical Specifications during power operation]. The changes eliminate testing that is likely to cause transients or excessive wear of equipment. An evaluation of these changes indicates that there will be a benefit to plant safety. The evaluation, documented in NUREG–1366, considered (1) unavailability of safety equipment due to testing, (2) initiation of significant transients due to testing, (3) actuation of engineered safety features that unnecessarily cycle safety equipment, (4) importance to safety of that system or component, (5) failure rate of that system or component, and (6) effectiveness of the test in discovering the failure.

As a result of the decrease in the testing frequencies, the risk of testing causing a transient and equipment degradation will be decreased, and the reliability of the equipment will not be significantly decreased.

The initial conditions and methodologies used in the accident analyses remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Therefore, accident analyses results are not impacted. Appropriate testing will continue to assure that equipment and systems will be capable of performing the intended function.

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

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

The proposed changes either modify allowable intervals between certain surveillance tests, delete surveillance requirements, or alter an action statement with regard to the required testing. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced by the revised surveillance or testing requirements.

Appropriate testing will continue to assure that the system is capable of performing its intended function. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, system, or component. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

All of the proposed technical specification changes are compatible with plant operating experience and are consistent with the guidance provided in GL 93–05 and NUREG–1366. The changes eliminate unnecessary testing that increases the risk of transients and equipment degradation. There is no impact on safety limits or limiting safety system settings.

The remaining proposed changes are administrative in nature and have no impact on the margin of safety of any technical specification. They do not affect any plant safety parameters or setpoints.

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

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

*NRC Project Director:* Robert A. Capra.

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

*Date of amendment request:* October 2, 1995

*Description of amendment request:* The proposed amendments would revise Section 3.4.2 to change the safety/relief valve (SRV) safety function lift setting tolerances from +1%, −3% to plus or minus 3% and include as-left SRV safety function lift setting tolerances of plus or minus 1%.

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

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

The probability of an accident previously evaluated will not increase as a result of this change, because the only changes are the tolerances for the SRV opening setpoints and the speed of the reactor core isolation cooling system (RCIC) turbine and pump. Changing the maximum allowable opening setpoint for the SRVs does not cause any accident previously evaluated to occur, or degrade valve or system performance in any way so as to cause an accident to occur with an increased frequency. In addition, the increased speed of the RCIC turbine and pump are within the design limits of the system. RCIC operability and failure probabilities are not impacted by this change.

The consequences of an ASME Overpressurization Event are not significantly increased and do not exceed the previously accepted licensing criteria for this event. General Electric (GE) has calculated the revised peak vessel pressure for LaSalle Station to be 1341 psig, which is well below the 1375 psig criterion of the ASME Code for upset conditions, referenced in Section 5.2.2, Overpressurization Protection, of the Updated Final Safety Analysis Report (UFSAR), and NUREG–0519 (Safety Evaluation Report related to the operation of LaSalle County Station, Units 1 and 2, March 1981), and Section 15.2–4, Closure of Main Steam Isolation Valves (BWR) of NUREG–0800 (Standard Review Plan).

GE has also performed an analysis of the limiting Anticipated Transient Without Scram (ATWS) event, which is the Main Steam Isolation Valve (MSIV) Closure Event. This analysis calculated the peak vessel pressure to be 1457 psig, which is sufficiently below the 1500 psig criterion of the ASME Code for emergency conditions.

Per NUREG–0519, listed above, Section 5.4.1, and Technical Specification 4.7.3.b, the RCIC pump is required to develop flow greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is supplied to the turbine at 1000 +20, −80 psig. Increasing the turbine and pump speed ensures these criteria will still be met and the consequences of an accident will not increase.

Therefore, there is not 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 only physical changes are to increase the allowable tolerances for SRV opening setpoints and to increase the RCIC pump and turbine speeds. These changes do not result in any changed component interactions. The SRVs and RCIC will still provide the functions for which they were designed. Since all of the other systems evaluated will continue to function as intended, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

While the calculated peak vessel pressures for the ASME Overpressurization Event and the MSIV closure ATWS Event are larger than that previously calculated without the proposed setpoint tolerance increases, the new peak pressures remain sufficiently below the respective licensing acceptance limits associated with these events. In addition, the actual L1C8 reload analysis of the ASME Overpressurization Event will be verified to be within the licensing acceptance limit for that event prior to Unit 1 Cycle 8 startup, as required in the normal reload 10 CFR 50.59 process. These licensing acceptance limits have been previously evaluated as providing a sufficient margin of safety. For other accidents and transients, the increased setpoint tolerances have a negligible effect on the results, so the margin of safety is preserved.

The staff has reviewed the amendment request and the licensee's no significant hazards consideration determination. Based on the review and the above discussions, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

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

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

*NRC Project Director:* Robert A. Capra.

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

*Date of amendment request:* October 17, 1995.

*Description of amendment request:* The proposed amendment would modify the Palisades Facility Operating License to reference 10 CFR Part 40, allow the use of source materials as reactor fuel, delete references to specific amendments and specific revisions in the listed titles of the Physical Security Plan Suitability Training and Qualification Plan and the Safeguards Contingency Plan, delete paragraph 2.F on reporting requirements, and make minor editorial changes. In addition, the Technical Specifications (TS) would be modified as follows: (1) TS 3.1.2 would be modified to change the pressurizer cooldown limit from 100 °F to 200 °F/hour; (2) the shield cooling system requirements would be relocated to the Palisades Final Safety Analysis Report (FSAR); (3) several minor editorial changes to various sections of the TS are proposed; and (4) revisions to several TS bases pages are proposed.

*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:

*Administrative Changes*

Since these changes have no effect on the physical plant or its operation, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

*Technical Changes*

The following evaluation supports the finding that operation of the facility in accordance with the two non-administrative changes would not:

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

Use of Source Material as reactor fuel: The use of depleted or natural uranium, defined as ''Source Material'' by 10 CFR 40.4, in addition to the currently allowed ''slightly enriched uranium'' would not affect the physical plant or its operation in any way which could increase the probability of any previously evaluated accident. Its use would not introduce any new kind or additional amount of fission product material. Therefore, use of source material as reactor fuel would not affect the consequences of an accident previously evaluated.

Restoration of the Pressurizer Cooldown Rate Limit: The Palisades Technical Specifications contain a single limit, item 3.1.2 b, for both heatup and cooldown rates for the pressurizer. The October 5, 1994 change request proposed changing that limit from 200°F/hour to 100°F/hour solely due to its inconsistency with the pressurizer design analysis. Fatigue calculations in the pressurizer design analysis assumed a heatup rate of 100°F/hour and a cooldown rate of 200°F/hour. Until issuance of Amendment 163, the Technical specifications contained a single limit for both heatup and cooldown rates of 200°F/hour. Although the installed equipment is not capable of exceeding the 100°F/hour heatup limit, the October 5, 1994 change request proposed a revised limit to assure that the Technical Specification limit was not less restrictive than the design analysis. The higher pressurizer cooldown rate does not affect the results of our analyses which determined the PCS Pressure-Temperature limits or the [Loss of Temperature Overpressurization] LTOP setting requirements of the Technical Specifications.

When the change was proposed, it was not realized that the more limiting cooldown rate might adversely, and unnecessarily, affect plant operation. This proposed change to the Technical Specifications would separate the limits for heatup rate and cooldown rate, returning the specified cooldown rate to the original value which was consistent with plant design. The current heatup rate limit, which is also consistent with the design, would be retained. The proposed pressurizer cooldown rate will allow depressurizing of the primary coolant system [PCS] and flooding the pressurizer steam space without undue restriction. The more rapid depressurization would be important in the event of a steam generator tube rupture.

Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would 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.*

Use of Source Material as reactor fuel: The use of depleted or natural uranium, defined as ''Source Material'' by 10 CFR 40.4, in addition to the currently allowed ''slightly enriched uranium'' would not affect the design (other than the fuel enrichment), configuration, or operation of the plant. Therefore this change cannot create the possibility of a new or different kind of accident from any previously evaluated.

Restoration of the Pressurizer Cooldown Rate Limit: The proposed change to the Technical Specifications would bring the plant within the assumptions of the design documents for the pressurizer and in line with the Accident analysis for the rapid reduction of the primary coolant system pressure. With the lower rate specified in the present technical specification, the depressurization of the PCS will be delayed to maintain the lower pressurizer cooldown rate.

Therefore, operation of the facility in accordance with the proposed change to the

Technical Specifications would 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.*

Use of Source Material as reactor fuel: The use of depleted or natural uranium, defined as ''Source Material'' by 10 CFR 40.4, in addition to the currently allowed ''slightly enriched uranium'' would not affect the Safety Limits, Limiting Conditions for Operation or other operating limits, or the safety analyses which they support. Therefore, the margin of safety is unaffected.

Restoration of the Pressurizer Cooldown Rate Limit: The proposed change to the Technical Specifications would bring the plant in line with the design analysis. This will not reduce the margin of safety since the higher rate is the basis for the present margin of safety.

Therefore, the proposed change to the Technical Specifications would 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:* Van Wylen Library, Hope College, Holland, Michigan 49423.

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

*NRC Project Director:* Brian E. Holian, Acting.

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

*Date of amendment request:* September 20, 1995.

*Description of amendment request:* The proposed amendment would allow a one-time extension of the 18-month surveillance intervals contained in the Technical Specifications (TS) related to system testing, instrumentation calibration, component inspection, component testing, response time testing and logic system functional tests for various systems, components and 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:

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

The proposed TS changes involve a one-time only change in the surveillance testing intervals to facilitate a one-time only change in the Fermi 2 operating cycle. The proposed TS changes do not physically impact the plant nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes do not significantly degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed TS changes affect only the frequency of the surveillance requirements and do not impact the TS surveillance requirements themselves. In addition, the proposed TS changes do not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the change in the frequency of surveillance testing. Also, the proposed TS changes do not significantly affect the availability of equipment or systems required to mitigate the consequences of an accident because of other, more frequent testing or the availability of redundant systems or equipment. Furthermore, a historical review of surveillance test results support the above conclusions. Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed TS changes involve a one-time only change in the surveillance testing intervals to facilitate the one-time only change in the Fermi 2 operating cycle. The propose TS changes do not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the surveillance test requirements themselves will remain unchanged. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Although the proposed TS changes will result in an increase in the interval between some surveillance tests, the impact, if any, on system availability is small based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any time dependent failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed TS changes do not significantly reduce a margin of safety.

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

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

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

*NRC Project Director:* Brian E. Holian, Acting.

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

*Date of amendment request:* August 8, 1995.

*Description of amendment request:* The amendments would revise Technical Specification Section 3/4.4.8, Table 4.4–4, Table Notations, to allow the reactor coolant system gross specific activity measurement method to be changed from the current degassed method to a non-degassed, or pressurized dilution, method.

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

*Criterion 1*

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. The amendments will have no effect on the probability of the occurrence of any accident. It has been demonstrated that the results obtained by the pressurized dilution technique are statistically similar to results obtained by the degassed technique. Therefore, implemention of the new method will have no effect insofar as the accuracy of the NC [reactor coolant system] system specific activity determination is concerned. Therefore, there will be no effect upon any accident dose consequences.

*Criterion 2*

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No accident causal mechanisms will be affected by installation of the sampling equipment required by the pressurized dilution technique. Operation of the NC system itself will not be affected by the proposed change in sampling technique. All procedure changes required for implementation of the new sampling method will be made according to the provisions of 10 CFR 50.59. No impact on other areas of plant operations will be generated as a result of the new sampling method.

*Criterion 3*

The requested amendments will not involve a significant reduction in a margin of safety. No impact on any safety limits will result from the change in sample method from the degassed technique to the pressurized dilution technique. Several benefits will result from the change, including fewer opportunities for valve mispositionings to occur, as well as reduced radiation exposure to Chemistry technicians. The proposed amendment is consistent with a similar amendment approved by the NRC for McGuire Nuclear Station (Amendment Nos. 66 and 47 for McGuire Units 1 and 2, respectively).

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

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

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

*Attorney for licensee:* Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

*NRC Project Director:* Herbert N. Berkow.

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

*Date of amendment request:* November 7, 1995.

*Description of amendment request:* The proposed change would revise Technical Specification 3/4.5.1 SAFETY INJECTION TANKS (SITs) by increasing the specified range associated with SIT water level and nitrogen cover pressure.

The current limiting conditions for operation (LCO) for the SIT requires that four SITs be operable with a water volume in the range of 1679 cubic feet (78%) to 1807 cubic feet (83.8%) and a nitrogen cover pressure between 600 psig to 625 psig. The proposed change requests an expanded range of 925.6 cubic feet (40%) to 1807 cubic feet (83.8%) for SIT level and 600 psig to 670 psig for SIT pressure indicators.

*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:

Operation of the facility in accordance with this change does not involve an increase in the probability of any accident. The SITs are used to mitigate the consequences of an accident and are not accident initiators.

The proposed change would actually decrease the consequence of events such as LOCA [loss of coolant accident] which would result in rapid RCS [reactor coolant system] depressurization.

By reducing SIT level, the initial nitrogen gas volume is increased which results in an increase in the SIT flow rate into the RCS for a given RCS pressure transient. This decreases the time required to fill the reactor vessel lower plenum after the end of blowdown. During refill, fuel cladding temperature increases rapidly due to insufficient cooling which is provided solely by rod to rod thermal radiation. Decreasing the refill time therefore, results in lower cladding temperature at the start of core reflood which results in lower Peak Cladding Temperature (PCT) during reflood.

Increasing the nitrogen cover pressure would also result in increased SIT flow rate and would be beneficial as described above.

Therefore, the proposed change will not involve a significant increase in the probability or consequence of any accident.

The proposed change will not create any new system connections or interactions. Thus, no new modes of failure are introduced. The increased range for SIT pressure and level is actually beneficial in maintaining lower PCT following a LOCA.

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

The impact of the proposed changes on the Waterford 3 FSAR [Final Safety Analysis Report] analyses have been evaluated. The AOR [Analysis of Record] shows that PCT and maximum cladding oxidation would increase slightly as a result of this change. However, they both remain below the acceptance criteria values of 2200 degrees fahrenhit and 17% for PCT and maximum cladding oxidation, respectively. The system capabilities to mitigate the consequences of accidents will be the same as they were prior to these changes.

Therefore, the proposed changes do[es] not involve a reduction in a margin of safety.

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

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

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

*NRC Project Director:* William D. Beckner.

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

*Date of amendment request:* August 10, 1995

*Description of amendment request:* This amendment would incorporate certain improvements into the Three Mile Island, Unit 1 Technical Specifications consistent with the Standard Technical Specifications for Babcock and Wilcox plants. The requested changes would affect the reactor building isolation instrumentation, sampling frequency for the sodium hydroxide tank, and the surveillance requirements for the plant vital bus batteries.

*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:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment involves changes to the TMI–1 Technical Specifications [TS] which are consistent with the [Babcock & Wilcox] B&W Standard Technical Specifications ([R]STS), NUREG–1430. This change does not involve any change to system or equipment configuration. The proposed amendment revises certain surveillance requirements, or extends certain surveillance intervals. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Therefore, this change does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The change only involves changes to surveillance requirements that are consistent with RSTS or deletion of requirements which are not appropriate for TS. No new failure modes are created and thus the changes are bounded by accidents previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. These proposed changes involve deletions of requirements or changes in surveillance requirements consistent with the B&W RSTS. No operating limits are affected and no reduction in the margin of safety is involved.

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

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

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

*NRC 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:* October 24, 1995.

*Description of amendment request:* The proposed amendment would revise the Technical Specification (TS) Surveillance Requirement of Section 4.4.5.1, ''Steam Generators'' and the Bases for Section 3/4.4.5, ''Steam Generators.'' Typographical errors in Section 4.4.5.1.3.c.1 and Table 4.4–6 are also proposed to be corrected. The proposed amendment would defer the next required surveillance to inspect steam generator tubes from October 20, 1996, to the next refueling outage or no later than October 20, 1997, whichever is earlier.

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

Pursuant to 10 CFR 50.92, NNECO [the licensee] has reviewed the proposed one-time change to extend the maximum allowable inspection interval for steam generator tubes from 24 months to 36 months. NNECO concludes that these changes do not involve a significant hazards consideration since the proposed change satisfies the criteria in 10 CFR 50.92(c). That is, the proposed changes do *not*:

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

This change involves one-time deferment of the eddy current inspection of the steam generator tubes until the end of the next refueling outage following the thirteenth fuel cycle, but no longer than 12 months beyond the original due date for the inspection. The steam generator tubes have only been exposed to one operating cycle and are made of thermally treated Alloy 690, one of the most corrosion resistant material currently used in recirculating steam generators. Following the first full fuel cycle of operation, the steam generator tube inspection found the tubes to be in excellent condition (i.e., no repairs were required and there was no evidence of an active degradation mechanism). Accordingly, no significant tube degradation is expected by the end of the thirteenth fuel cycle. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

This one-time change, allowing the steam generator tubes to be examined at the end of the refueling outage following Cycle 13 does not alter the physical design, configuration, or method of operation of the plant. The extension of the inspection interval is not expected to result in significant steam generator tube degradation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Steam generator tube degradation occurs primarily during operation. The change to extend the maximum allowable inspection interval for steam generator tubes from 24 months to 36 months will not significantly increase the total operating time during Cycle 13 (the plant was in an outage for at least 10 months of the 12 month extension). Therefore, there is no significant effect on the extent and severity of tube degradation. The improved corrosion resistance of the steam generators tubes (thermally treated Alloy 690) minimizes the threat of primary- and secondary-side corrosion. No indications of corrosion have been identified in inspections performed so far. Based on our assessment of the inspection data and corrosion potential, all tubes are expected to be within the Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' limits by the end of Cycle 13. Also, correction of the typographical errors will improve the fidelity of the specification. Therefore, this change does not involve a significant reduction in the margin of safety.

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

*Local Public Document Room location:* 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:* May 1, June 14 and 29, July 14, 17, 18, and 26, 1995 with supplemental information provided by letter dated October 20, 1995.

*Description of amendment request:* Each proposed amendment would change the surveillance requirement frequency from the current once per 18-month interval to once per 24-month which is the current length of a Millstone Unit 3 refueling cycle. The changes pertain to the following equipment:

May 1, 1995, Flow Paths—Operating; Position Indication System; Rod Drop Time; Seismic Monitoring System; Loose Part Detection System; Quench Spray System; Containment Recirculation Spray System; Containment Isolation Valves. This notice supersedes the notice published in the Federal Register on June 6, 1995 (60 FR 29882) relating to containment isolation valves.

May 1, 1995, Steam Generator Tube Inspections; 10CFR50, Appendix J, Type B and Type C Tests.

June 14, 1995, AC Sources Operating; DC Sources Operating; Containment Penetration Conductor Overcurrent Protective Devices; Motor-Operated Valves Thermal Overload Protection.

June 29, 1995, Electric Hydrogen Recombiners; Auxiliary Feedwater System; Reactor Plant Component Cooling Water System; Service Water System; Snubbers.

July 14, 1995, ECCS Subsystems—Tavg Greater Than or Equal to 350 °F; pH Trisodium Phosphate Storage Baskets.

July 17, 1995, Supplementary Leak Collection and Release System; Control Room Emergency Ventilation System; Control Room Envelope Pressurization System; Auxiliary Building Filter System; Fuel Building Exhaust Filter System.

July 18, 1995, Reactor Coolant System.

July 26, 1995; Reactor Trip System Instrumentation; ESFAS Instrumentation; Remote Shutdown Instrumentation; Accident Monitoring Instrumentation; RCS Total Flow Rate; Process and Radiation Monitoring Instrumentation.

In addition, the specifications are changed from a five-column to a one-column format.

*Basis for proposed no significant hazards consideration determination:* The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The proposed changes to surveillance requirements of the Millstone Unit No. 3 Technical Specifications extend the frequency for checking the operability of the affected components/equipment. The proposal would extend the frequency from at least once per 18 months to at least once each refueling interval (i.e., nominal 24-months).

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

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

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

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

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

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

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

The proposed changes to surveillance requirements of the Millstone Unit No. 3 Technical Specifications extend the frequency for verifying the operability of the affected components/equipment. The proposal would extend the frequency from at least once per 18 months to at least once each refueling interval (nominal 24 months).

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

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

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

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

The proposed changes to surveillance requirements of the Millstone Unit No. 3 Technical Specifications extend the frequency for verifying the operability of the components/equipment. The proposal would extend the frequency from at least once per 18-months to at least once each refueling interval (24-months).

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

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

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

*Local Public Document Room location:* 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:* September 29, 1995.

*Description of amendment requests:* The amendments would add a one-time footnote to the Technical Specifications regarding the emergency diesel generator diesel fuel oil storage and transfer system to permit the existing storage tanks to be replaced with double walled tanks and piping that comply with new California regulations.

*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.

Neither the emergency diesel generators (EDGs) nor the diesel fuel oil (DFO) storage and transfer system is an accident initiator. When performing the modifications to the

DFO storage tanks and transfer piping, administrative compensatory measures will be taken to reduce the potential challenge to the EDGs and to verify the operability of the DFO transfer system. A probabilistic risk assessment (PRA) was performed and demonstrates that the change in core damage frequency associated with taking each DFO storage tank and its associated suction transfer piping out of service for 60 days (total of 120 days for both trains) is not significant considering the compensatory measures which will be taken during the tank replacement period.

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.

Neither the EDGs nor the DFO storage and transfer system is an accident initiator. Temporary DFO storage will be onsite during tank replacement. The fire protection guidelines in Appendix 9.5B of the Updated Final Safety Analysis Report will be complied with in order to ensure temporary DFO storage without risk to plant systems.

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

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

The proposed changes considering implementation of the compensatory measures has been shown to not impair safe operation of the plant. Having one DFO storage tank and associated piping out of service does not reduce the margin of safety since temporary storage of DFO will be maintained onsite and administrative compensatory measures will be taken to minimize the potential impact of this condition. Additionally, delivery of DFO to the site is available within 24 hours if needed.

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.

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:* October 4, 1995.

*Description of amendment requests:* The amendments would relocate the requirements in ten sub-sections of the Technical Specifications to licensee controlled documents in accordance with the guidance in the Commission's Final Policy Statement and the Commission's revisions to 10 CFR 50.36 (60 FR 36959, July 19, 1995) on the content of Technical Specifications and the Standard Technical Specifications, Westinghouse Plants, NUREG–1431, Rev. 1, dated April 1995. The ten sub-sections which the licensee proposes to relocate, without changes to the requirements, to the Updated Final Safety Analysis Report or other controlled documents relate to: boration system flow path, position indication system, rod drop time, seismic instrumentation, chlorine detection system, turbine overspeed protection, containment leakage, containment structural integrity, electrical equipment protective devices and containment penetration conductor overcurrent protective devices.

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

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

The proposed changes simplify the Technical Specifications (TS), meet regulatory requirements for relocated TS, and implement the recommendations of the Commission's Final Policy Statement on TS Improvements and revised 10 CFR 50.36. Future changes to these requirements will be controlled by 10 CFR 50.59. The proposed changes are administrative in nature and do not involve any modifications to any plant equipment or 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 changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety-related system performs its function. Also, no changes to the operation of the plant or equipment are involved.

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

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

The proposed changes involve relocating TS requirements to a licensee-controlled document. The requirements to be relocated were identified by applying the criteria endorsed in the Commission's Final Policy Statement, which is included in the new revision of 10 CFR 50.36, and are consistent with NUREG–1431, Rev. 1 (Reference 2). Thus, the proposed changes do not alter the basic regulatory requirements and do not affect any safety analysis.

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.

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

*Date of amendment request:* November 2, 1995.

*Description of amendment request:* The proposed amendment would revise Section 5.0, Administrative Controls, of the Trojan Nuclear Plant Technical Specifications, Appendix A to License NPF–1, to reflect changes in the organization of the Portland General Electric Company (PGE) as they apply to oversite and management of the Trojan Nuclear Plant.

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

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

The changes in management titles and reporting relationships are administrative in nature, do not alter the intent of the Possession Only License, and do not modify the present plant systems or adminstrative controls necessary to preserve and protect the integrity of the nuclear fuel at the Trojan Nuclear Plant. The Trojan Site Executive and Plant General Manager will be located at the site and will continue to provide senior management attention to each of the functional areas in the Trojan Nuclear Plant organization during decommissioning of the facility.

The general classification of accidents for the permanently defueled condition are limited. The three classifications are (1) radioactive release from a subsystem or component, (2) fuel handling accident, and (3) loss of spent fuel decay heat removal capability. The probability of occurrences of consequences from these accidents remain unchanged and are bounded by the current accident analysis. Therefore, the requested changes do not involve a significant increase in the probability or occurrence of an accident previously evaluated.

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

The requested amendment is administrative in nature, does not affect the manner in which systems and components are operated or maintained, and does not alter the intent of the Possession Only License. The accident scenarios associated with the permanently defueled condition are limited to (1) radioactive release from a subsystem or component, (2) fuel handling accident and (3) loss of spent fuel decay heat removal capability. There are no new accident scenarios or failure modes created by the requested administrative changes. Therefore the requested change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The requested amendment is administrative in nature, does not affect the manner in which systems and components are operated or maintained, does not alter the intent of the Possession Only License, nor does it adversely impact previously accepted margins of safety. Therefore, the requested amendment does not involve a significant reduction in margin of safety.

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

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

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

*NRR Project Director:* Seymour H. Weiss.

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

*Date of amendment request:* October 7, 1995 as supplemented by letter dated October 27, 1995.

*Description of amendment request:* The proposed change to Hope Creek Technical Specifications (TSs) 4.8.1.1.2, ''A.C. Sources—Operating'', would replace the reference to a voltage and frequency band for the 10 second starting time test with a minimum required voltage and frequency that must be attained within 10 seconds.

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

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

Since no change is being made to the offsite power supplies, or to any system or component that interfaces with the offsite power supplies, there is no change in the probability of a Loss of Offsite Power Accident.

Since the proposed change still ensures the surveillance requirements meet the licensing basis and since the full spectrum of loading, unloading and standby testing performed at the 18 month frequency continues to demonstrate the capability of the diesel generators to satisfy onsite power requirements during simulated accident conditions while the monthly testing demonstrates availability, there is no change in the consequences of an accident.

Since the proposed change will eliminate unnecessary adjustments to the governor controls, the probability of malfunction is potentially reduced.

This change ensures the surveillance requirements reflect the design basis and provide a basis for consistent timing methodology. Since the proposed change is consistent with the intent of the existing specifications, and with the design basis of the system and since no physical changes are being proposed, no action will occur that will increase the probability or consequences of an accident or malfunction of equipment important to safety. The diesel generators will continue to function as stated in the UFSAR [Updated Final Safety Analysis Report].

Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated.

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

The proposed change does not result in any design or physical configuration changes to the offsite power supplies or to the diesel generators. Operation in accordance with the proposed change will not impair the diesel generators ability to perform as provided in the design basis. By eliminating unnecessary adjustments to the diesel generator governor control, performance during any accident is potentially enhanced. The diesel generators will continue to function as stated in the UFSAR. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

Since the proposed change does not involve the addition or modification of plant equipment, is consistent with the intent of the existing Technical Specifications, meets the intent of applicable Regulatory Guides, and is consistent with the design basis of the diesel generators and the UFSAR, no action will occur that will 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:* Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

*NRC Project Director:* John F. Stolz.

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

*Date of amendment request:* September 29, 1995.

*Description of amendment request:* The proposed amendment would modify Technical Specification (TS) 3/4.4.3, Safety Valves and Pilot Operated Relief Valve—Operating, and associated Bases 3/4.4.2 and 3/4.4.3, Safety Valves, to increase the lift setting of the pressurizer code safety valves (PSVs) to [equal to or less than] 2575 psig, which corresponds to a lift setting tolerance of +3% of the nominal lift pressure. Increasing the upper bound of the lift setting tolerance of the PSVs from +1% to +3% will allow normal surveillance testing of the PSVs to be within +3% of the nominal lift setpoint of 2500 psig, which is still acceptable.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because increasing the PSV lift tolerance from +1% to +3% only affects the as-found tolerance of the PSVs. The initial setting tolerance will still be limited to +1%. No hardware modification will be done to the valves which could affect any accident initiators.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because increasing the PSV lift tolerance from +1% to +3% does not affect the radiological releases of any accident previously evaluated in the [Updated Safety Analysis Report] USAR. This is not a hardware modification and the reactor coolant pressure boundary integrity is unaffected.

2. Not create the possibility of a new kind of accident from any previously evaluated because increasing the PSV lift tolerance from +1% to +3% allows the PSVs to protect the reactor coolant pressure boundary from overpressure transients. This change only affects the allowable lift tolerance. The initial lift setting tolerance is still less than +1%. This change does not modify the valve hardware or alter the operation of the valves. The possibility of the valves spuriously opening during power operation will not be changed. The valve setpoint with a −3% lift tolerance is well above the normal operating conditions and the [reactor coolant system] RCS high pressure trip setpoint.

3. Not involve a significant reduction in a margin of safety because at the +3% lift tolerance the RCS pressure and the reactor thermal power are still within the USAR acceptance criteria for a control rod withdrawal at low power. This change ensures the Technical Specification lift setpoint tolerances are consistent with the requirements given in the [American Society of Mechanical Engineers] ASME Boiler and Pressure Vessel Code.

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

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

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

*NRC Project Director:* Gail H. Marcus.

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

*Date of amendment request:* June 21, 1994, as amended by letter dated October 23, 1995.

*Description of amendment request:* The proposed amendment would relocate the review and audit requirements of the On-site Review Committee (ORC) and Nuclear Safety Review Board (NSRB) contained in TS 6.5.1, TS 6.5.2 and TS 6.5.3 to the Operational Quality Assurance Manual (OQAM). In addition, the proposed amendment would delete reference to the Manager, Nuclear Safety and Emergency Preparedness in TS 6.2.3. A revision to the Index was proposed to reflect the relocations. This amendment request was previously published in the Federal Register on August 31, 1994 (59 FR 45036).

*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 changes are administrative and equivalent descriptions and requirements for these oversight committees are contained in the OQAM.

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

These changes do not involve any physical alterations to the plant. There is no new type of accident or malfunction created and the method and manner of plant operation will not change. The changes are administrative and equivalent descriptions and requirements for these oversight committees are contained in the OQAM.

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

The margin of safety remains unaffected since no design change is made and plant operation remains the same. The changes are administrative and equivalent descriptions and requirements for these oversight committees are contained in the OQAM.

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

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

*NRC Project Director:* William H. Bateman.

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

*Date of amendment request:* October 17, 1995.

*Description of amendment request:* The proposed change would revise the Technical Specifications (TS) for the North Anna Power Station, Unit No. 2 (NA–2). Specifically, the proposed change would reduce from two to one the minimum number of steam generators (SGs) required to be opened for inspection during the first refueling outage following an SG replacement. TS surveillance requirements 4.4.5.0 through 4.4.5.5 for inspection of the SG tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. Accordingly, the purpose of TS 4.4.5.1 is to require periodic sample inspections of SGs. The initial inspection after SG replacement combined with the subsequent inservice inspections serve to provide reasonable assurance of detection of structural degradation of the tubes. The proposed TS change does not affect or change this basis. However, the requirement that two SGs would be opened and inspected during the first refueling outage after SG replacement is considered unnecessary.

The NA–2 SGs were replaced during the first quarter of 1995. The purpose of SG replacement was to restore the integrity of the SG tubes to a level equivalent to new SGs. In reality, replacement SG components incorporate a large number of design improvements which reflect the ''state-of-the-art'' technology that currently exists for SG design. These design improvements will improve the long-term maintainability and reliability of the replacement SGs. These enhancements do not adversely affect the mechanical or thermal-hydraulic performance of the SGs. Thus, the replacement SGs are considered superior to the original SGs in terms of design and materials.

The proposed TS change does not affect or change any limiting conditions for operation (LCO) or any other surveillance requirements in the TS and the Basis for the surveillance requirement remains unchanged. An inspection of the minimum required number of tubes will still be performed prior to returning the SGs to service. Although the proposed change reduces the number of SGs required to be opened for inspection, the minimum number of tubes required to be examined during the inspection is not being changed. Thus, the minimum inspected tube population size would not be changed.

*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:

We have evaluated the proposed change against the criteria described in 10 CFR 50.92 and concluded that the proposed Technical Specifications change does not pose a significant hazards consideration.

[1] The proposed Technical Specifications change does not affect the assumptions, design parameters, or results of any UFSAR [Updated Final Safety Analysis Report] accident analysis and the proposed amendment does not add or modify any existing equipment. Therefore, the proposed Technical Specifications change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

[2] The proposed change to the Technical Specifications does not involve modifications to any of the existing equipment or affect the operation of any existing systems. The absence of any hardware or software changes means that the accident initiators remain unaffected, so no unique accident possibility is created. Therefore, the proposed Technical Specifications change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

[3] Although the proposed change will reduce the minimum number of steam generators required to be opened for inspection during the first refueling outage following steam generator replacement, the revised Technical Specification surveillance will continue to ensure that a sampling of steam generator tubes will be inspected. The operability of the steam generators will also continue to be verified by periodic inservice inspections. Therefore, since equipment reliability will be maintained, the proposed Technical Specifications change will not involve a significant reduction in a margin of safety.

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

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

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

*NRC Project Director: David B. Matthews.*

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

*Date of amendment request: October 18, 1995.*

*Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) 3.4, ''Steam and Power Conversion System,'' by modifying and clarifying the operability requirements for the main steam safety valves (MSSVs), the auxiliary feedwater (AFW) System, and the condensate storage tank system.*

The proposed amendment would eliminate inconsistencies within TS Section 3.4 and provide the basis for acceptable operation of the Auxiliary Feedwater System below 15% reactor power. The proposed amendment supersedes in its entirety a previously submitted proposed amendment dated May 20, 1994, which was noticed in the Federal Register on September 28, 1994 (59 FR 49442).

*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:

*Significant Hazards Determination for Proposed Changes to Technical Specification (TS) 3.4.a ''Main Steam Safety Valves''*

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

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

Currently, TS 3.4.a.1.A.2 requires five MSSVs to be operable prior to heating the reactor > 350 °F. The proposed change requires a minimum of two MSSVs per steam generator to be operable prior to heating the reactor coolant system > 350 °F, and five MSSVs per steam generator to be operable prior to reactor criticality. If these conditions cannot be met within 48 hours, within 1 hour action shall be initiated to achieve hot standby within 6 hours, achieve hot shutdown within the following 6 hours, and achieve and maintain the reactor coolant system temperature < 350 °F within an additional 12 hours.

The MSSVs are relied upon to function in each of the following USAR analyzed accidents: Reactor Coolant Pump Locked Rotor, Loss of External Electrical Load, Loss of Normal Feedwater, Uncontrolled Rod Cluster Control Assembly Withdrawal, Steam Generator Tube Rupture, and Anticipated Transients without Scram.

In a subcritical condition, two operable MSSVs are capable of relieving the maximum steam generated during these anticipated design basis transient events. Because this proposed TS requires all MSSVs to be operable prior to reactor criticality, there will be no adverse effect on the health and safety of the public.

In all cases, the relieving capacity of the MSSVs is sufficient to maintain steam pressures within safety analysis acceptable criteria, and reactor criticality is not permitted unless all MSSVs are operable. Therefore, there is no adverse effect on the health and safety of the public and no significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The USAR safety analysis assumes five MSSVs per steam generator are operable. However, as shown above, this change results in no steam generator overpressure event or increase in the radiological dose. Therefore, this change will not involve a reduction in the margin of safety.

*Significant Hazards Determination for Proposed Changes to Technical Specification (TS) 3.4.b ''Auxiliary Feedwater System''*

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

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

Current TS 3.4.a.1.A.1 and TS 3.4.b governing auxiliary feedwater flow to the steam generators are being combined and titled, ''Auxiliary Feedwater System.'' This change is consistent with the format of ''Westinghouse Standard Technical Specifications,'' NUREG–1431. In addition to the formatting changes, a number of technical changes are being proposed. These are:

The correction of an inconsistency between current TS 3.4.a.1.A.1 and current TS 3.4.b.2.A.

The addition of a seven (7) day Limiting Condition for Operation (LCO) action statement for one inoperable steam supply to the turbine driven auxiliary feedwater pump.

A specification is being added to permit any of the following conditions with reactor power less than 15%, without declaring the corresponding AFW train inoperable: the AFW pump control switches located in the control room to be in the ''pullout'' position, flow control valves AFW–2A and AFW–2B to be in a throttled or closed position, and train cross-connect valves AFW–10A and AFW–10B to be in the closed position.

An inconsistency currently exists between current TS 3.4.a.1.A.1 and current TS

3.4.b.2.A. TS 3.4.a.1.A.1 requires the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators to be operable, with a corresponding 48 hour limiting condition for operation (LCO) action statement if this requirement is not met. TS 3.4.b.2.A allows one auxiliary feedwater pump to be inoperable for 72 hours. This arrangement can cause a conflict regarding which TS is applicable depending on which component in the auxiliary feedwater flowpath to the steam generators is inoperable. By moving all TS action statements to TS 3.4.b, the inconsistency between TS 3.4.a.1.A.1 and TS 3.4.b.2.A will be eliminated. The requirement to maintain the operability of the system piping and valves directly associated with providing auxiliary feedwater flow to the steam generators remains, but is being modified to prevent the removal of both AFW supply headers from service.

Proposed TS 3.4.b.2.C is being added to allow one steam supply to the turbine driven auxiliary feedwater pump to be inoperable for seven days. This addition is consistent with ‘‘Westinghouse Standard Technical Specifications,’’ NUREG–1431. The seven day completion time is reasonable based on the redundant steam supplies to the pump, the availability of the redundant motor-driven AFW pumps, and the low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump. For these reasons, this change will have no adverse effect on the health and safety of the public.

Proposed TS 3.4.b.6.A and B permit the AFW Pump control switches located in the control room to be placed in the ‘‘pull out’’ position and valves AFW–2A and AFW–2B to be in a throttled position when below 15% reactor power without declaring the corresponding AFW train inoperable. This change is proposed to resolve concerns regarding the cycling of the AFW pumps and the throttling of valves AFW–2A and AFW–2B during plant startups and shutdowns. Analysis shows that control room operators have a minimum of ten minutes to initiate auxiliary feedwater flow after a design basis accident with no steam generator dryout or core damage.

All accidents which rely on AFW flow for mitigation were reanalyzed to support this change. These analyses were completed assuming an initial power of 100%. However, a 15% reactor power restriction has been imposed on placing the AFW pump control switches located in the control room in the ‘‘pull out’’ position and throttling valves AFW–2A and AFW–2B. This restriction in effect limits use of TS 3.4.b.6 to plant startups, shutdowns and other low power operating conditions.

This change alters the assumptions of the safety analysis for the Small-Break Loss of Coolant Accident, the Steam Generator Tube Rupture and the Loss of Normal Feedwater due to their dependence on the AFW system to start and supply AFW for heat removal. To support this change, the Westinghouse Electric Corporation performed an analysis of the Small-Break Loss-of-Coolant Accident using the NOTRUMP code assuming ten minutes for operator action to initiate auxiliary feedwater. This analysis resulted in a Peak Cladding Temperature (PCT) of 1053 °F from an initial power level of 100%. In addition, all other acceptance criteria of 10 CFR 50.46 were met. This large margin to the 2200 °F PCT limit supports ten minutes for operator action to initiate auxiliary feedwater.

Furthermore, WPSC has analyzed the Loss of Normal Feedwater and the Steam Generator Tube Rupture Accident assuming delays in the initiation of auxiliary feedwater. The Loss of Normal Feedwater Accident with a ten minute delay in the initiation of Auxiliary Feedwater does not result in any adverse condition in the core. It does not result in water relief from the pressurizer safety valves, nor does it result in uncovering the tube sheets of the steam generators. Also, at all times the Departure from Nucleate Boiling Ratio (DNBR) remained greater than 1.30. The Steam Generator Tube Rupture Accident with no auxiliary feedwater flow was also analyzed. The results of this analysis indicate that neither steam generator empties of liquid and at least 20 °F of reactor coolant system subcooling is maintained throughout the transient. Also, there is no increase in the radiological dose to the public.

Ten minutes is an acceptable time for operator action because four independent alarms in the control room would initiate operator action to place the AFW pump control switches to the ‘‘auto’’ position and initiate AFW flow to the steam generators when necessary. These include two steam generator lo level alarms (one per steam generator), and two steam generator lo-lo level alarms (one per steam generator). Provisions also exist to add additional low level alarms on the plant process computer. In addition to these alarms, control room operators have twelve other indications of insufficient, or no, AFW flow to the steam generators. These indications include three auxiliary feedwater pump low discharge pressure alarms (one per AFW pump), two auxiliary feedwater flow meters (one per steam generator), two AFW pump motor amp meters (one per motor-driven AFW pump), two ‘‘ESF in Pullout’’ alarms (one per Engineered Safety Features train) and three pump running lights (one per AFW pump). The ten minutes for operator action was discussed in a telephone conversation between WPSC and Mr. R. Laufer (NRR). Ten minutes for operator action is further supported by Branch Technical Position EISCB 18. Scenarios have been completed on the KNPP simulator to support ten minutes for operator initiation of AFW flow. In all cases, operators manually initiated AFW flow within the allowed ten minutes.

Proposed TS 3.4.b.6.C permits valves AFW–10A and AFW–10B to be in the closed position when below 15% reactor power without declaring the turbine-driven AFW train inoperable. This change is being proposed to allow operational flexibility of the AFW system during startups and shutdowns. As described below, the operability of the turbine-driven auxiliary feedwater train is independent of the position of the valves AFW–10A and AFW–10B. However, the operability of this train is dependent on the ability of these valves to reposition.

The operability of the AFW system following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80–04. As a result of this review, requirements for the turbine-driven AFW pump were originally added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW–10A and AFW–10B) are normally maintained in the open position. This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally open or closed. Therefore, the position of the cross-connect valves does not affect the operability of the turbine-driven AFW train. However, operability of the train is dependent on the ability of the valves to reposition.

For these reasons, this change will have no adverse effect on the health and safety of the public or significantly increase the probability or consequences of an accident previously evaluated in the USAR.

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

The auxiliary feedwater system is required to mitigate the consequences of an accident. The auxiliary feedwater system is not an accident initiator. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

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

This change alters the assumptions of the safety analysis for the Small-Break Loss-of-Coolant Accident, the Steam Generator Tube Rupture and the Loss of Normal Feedwater due to their dependence on the AFW system to start and supply AFW flow for heat removal. To support this change the Westinghouse Electric Corporation has performed an analysis of the Small-Break Loss-of-Coolant Accident using the NOTRUMP code assuming ten minutes for operator action to initiate auxiliary feedwater. This analysis resulted in a Peak Cladding Temperature (PCT) of 1053° F from an initial power level of 100%. In addition, all other acceptance criteria of 10 CFR 50.46 were met. This large margin to the 2200° F PCT limit supports ten minutes for operator action to initiate auxiliary feedwater.

Furthermore, WPSC has analyzed the Loss of Normal Feedwater and the Steam Generator Tube Rupture Accident assuming delays in the initiation of auxiliary feedwater. The Loss of Normal Feedwater Accident with a ten-minute delay in the initiation of Auxiliary Feedwater does not result in any adverse condition in the core.

It does not result in water relief from the pressurizer safety valves, nor does it result in uncovering the tube sheets of the steam generators. Also, at all times the Departure from Nucleate Boiling Ratio (DNBR) remained greater than 1.30. The Steam Generator Tube Rupture Accident with no Auxiliary Feedwater flow was also analyzed. The results of this analysis indicate that neither steam generator empties of liquid and at least 20° F of reactor coolant system subcooling is maintained throughout the transient. Also, there is no increase in the radiological dose to the public. For these reasons, these changes will not adversely affect the health and safety of the public or involve a significant reduction in the margin of safety.

As discussed in the safety evaluation, the operability of the turbine-driven AFW train is independent of the position of valves AFW–10A and AFW–10B. However, the operability of the train is dependent on the ability of these valves to be repositioned. Therefore, the proposed change has no impact on the accident analysis and no effect on the margin of safety.

*Significant Hazards Determination for Proposed Administrative Changes to Section TS 3.4, ''Steam and Power Conversion System''*

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

1. Involve a significant increase in the probability or 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 the margin of safety.

The proposed changes are administrative in nature and do not alter the intent or interpretation of the TS. Therefore, no significant hazards exist.

Additionally, the proposed change is similar to example C.2.e(i) in 51 FR 7751. Example C.2.e.(i) states that changes which are purely administrative in nature; i.e., to achieve consistency throughout the Technical Specifications, correct an error, or a change in nomenclature, are not likely to involve a significant hazard.

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

*Local Public Document Room location:* University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311–7001.

*Attorney for licensee:* Bradley D. Jackson, Esq., Foley and Lardner, PO Box 1497, Madison, Wisconsin 53701–1497.

*NRC Project Director:* Gail H. Marcus.

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

*Date of amendment request:* October 18, 1995.

*Description of amendment request:* This license amendment would replace the current fuel oil volume requirement in the emergency diesel generator (EDG) day tank in Technical Specifications 3.8.1.1.b.1) and 3.8.1.2.b.1) with a fuel oil level requirement. Associated Surveillance Requirement 4.8.1.1.2.a.1) would also be changed to replace the requirement to visually check the fuel oil level in the day tank with a requirement to verify that the fuel oil transfer pump starts on low level in the day tank standpipe.

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

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

The proposed change will increase the minimum amount of diesel fuel oil that the current specifications require to be maintained in the EDG day tanks for standby operation. This change reflects the level that has been administratively maintained since the beginning of plant operation. The proposed change will not affect the way the EDG is operated and does not affect the ability of the EDGs to perform their safety function. The surveillance requirement change is being made to more thoroughly reflect the method used to assure the tank level is being properly maintained. The proposed change will not require the EDG to be operated in a manner different than that for which it was designed. Therefore, the proposed change will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

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

There are no active components being added whose failure could prevent the EDG from functioning. There is no new type of accident or malfunction being created and the method and manner of plant operation remains unchanged. The safety design bases in the USAR have not been altered. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. The method of operation of the EDGs is not being altered, and the fuel oil transfer pumps will continue to perform the same function they currently perform. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

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

There are no changes being made to any safety limits or safety system settings that would adversely impact plant safety. Although the minimum required amount of fuel oil specified in the Technical Specifications is being revised, this amount of fuel oil has been administratively controlled since the beginning of commercial operation. Thus, the operability of the emergency diesel generators has never been affected by this issue. Neither the method of operation of the EDGs nor their safety function are being altered by the proposed change. Therefore, the proposed change would not result in a reduction in a margin of safety.

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

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

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

*NRC Project Director:* William H. Bateman.

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

*Date of amendment request:* October 24, 1995.

*Description of amendment request:* This license amendment request proposes to revise Surveillance Requirement 4.7.6.e.4 to reflect a design change, scheduled to be installed during the next refueling outage, that would change the output rating of the charcoal filter adsorber unit heater in the pressurization portion of the control room emergency ventilation system (CREVS) from 15 kW to 5 kW. Proposed revisions to Surveillance Requirements 4.7.6.c.2 and 4.7.6.d are included which would change the acceptance criteria for the testing of carbon samples from the CREVS charcoal adsorbers. The proposal would adapt ASTM D 3803–1989 as the laboratory testing standard with the testing to be performed at 30 degrees Centigrade and 70 percent relative humidity for a methyl iodide penetration of 2 percent.

*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 design function of the filter adsorber unit heater in the pressurization system portion of CREVS is to reduce the relative humidity of the air entering the charcoal filter beds to 70% relative humidity. Although the original design specified a heater with a rating of 15 kW, review of the design basis calculation for this system indicates that only 2.09 kW is actually required (including applicable margins to allow for voltage variations). The proposed change to the CREVS heaters' output rating from 15 kW to 5 kW will not affect the method of operation of the system, and the new heater capacity will still exceed filter operational requirements and safety margin. Neither the heater change nor the charcoal testing protocol changes will affect system operation or performance, nor do they affect the probability of any event initiators. These changes do not affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. Therefore, the proposed changes will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

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

The requested change to the CREVS heaters' output rating and the changes to the charcoal sample testing protocol will not affect the method of operation of the system, and the new heater capacity will still exceed filter operational requirements and safety margin by a significant amount. The proposed changes only affect the heater size in the system and the testing criteria for the charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

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

The requested change to the CREVS heaters' output rating will reduce the heater output of the system, but the new heater capacity will still exceed filter operational requirements and safety margin by a significant amount. In addition, the reduction in heat load output from the heater will increase the design margin between the cooling capacity of the system air conditioning units and the building heat load. The new charcoal adsorber sample laboratory testing protocol is more stringent than the current testing practice and more accurately demonstrates the required performance of the adsorbers following a design basis LOCA [loss-of-coolant accident]. Therefore, these changes will not reduce the margin of safety of the CREVS filter operation.

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

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

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

*NRC Project Director:* William H. Bateman.

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

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

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

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

*Date of amendment request:* September 20, 1995.

*Description of amendment request:* The proposed amendment would modify the Appendix A Technical Specifications for the Engineered Safety Features Actuation System (ESFAS) Instrumentation. Specifically, the proposed amendment would revise the Seabrook Station Technical Specifications to relocate Functional Unit 6.b, ''Feedwater Isolation—Low RCS $T_{avg}$ Coincident with a Reactor Trip'' from Technical Specification 3.3.2. ''Engineered Safety Features Actuation System Instrumentation'' to the Seabrook Station Technical Requirements Manual which is a licensee controlled document.

*Date of publication of individual notice in* Federal Register: October 24, 1995 (60 FR 54524).

*Expiration date of individual notice:* November 24, 1995.

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

### *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.

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

*Date of application for amendments:* October 25, 1994, as supplemented by letter dated September 11, 1995.

*Brief Description of amendments:* The proposed amendments change the Technical Specifications to relocate the remaining Environmental Technical Specifications to other licensee-controlled documents and delete the 30-day reporting requirement for inoperable meteorological instrumentation.

*Date of issuance:* November 2, 1995.

*Effective date:* November 2, 1995.

*Amendment Nos.:* 179 and 210.

*Facility Operating License Nos. DPR–71 and DPR–62.* Amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: December 7, 1994 (59 FR 63113). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* May 13, 1993 as supplemented August 11 and September 20, 1995.

*Brief description of amendments:* The amendments revised Section 3/4.6.1.7 of the Technical Specifications, Containment Purge Ventilation System, to allow the simultaneous opening of the 8-inch miniflow purge supply and exhaust valves to ensure the containment atmosphere is conducive to human occupants and to maintain their dose as low as reasonably achievable.

*Date of issuance:* November 2, 1995.

*Effective date:* November 2, 1995.

*Amendment Nos.:* 76, 76, 68, and 68.

*Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77:* The amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: September 15, 1993 (58 FR 48379). The August 11 and September 20, 1995, submittals provided clarifying information that 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 2, 1995.

No significant hazards consideration comments received: No

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

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

*Date of application for amendments:* September 1, 1995, as supplemented on September 1 (two letters), September 2, September 4, September 8, September 15, September 19, September 20, September 22, October 3, October 7, October 11 (two letters), October 13 (three letters), October 23 and October 26, 1995.

*Brief description of amendments:* The amendments revise the steam generator (SG) repair criteria in the Byron, Unit 1 and Braidwood, Unit 1 Technical Specifications. These revisions add a set of voltage-based SG tube repair criteria different from those previously added by License Amendment No. 66, dated October 24, 1994, to the Byron 1 TSs and by License Amendment No. 54, dated August 18, 1994, to the Braidwood 1 TSs. The present set of voltage repair limits which are being added to the Byron 1 and Braidwood 1 TSs are applicable only for a specific form of SG tube degradation identified as outer diameter stress corrosion cracking (ODSCC) which is confined entirely within the thickness of the tube support plates (TSPs) in the SGs. The voltage-based repair criteria for the cold-leg side of the SGs for SG tubes with ODSCC indications and for SG tubes on the hot-leg side which show significant denting, are consistent with those provided in the NRC staff's guidance contained in Generic Letter 95–05, dated August 3, 1994.

The lower voltage repair limit for the SG tubes with ODSCC indications on the hot-leg side of the SGs have been raised from 1.0 to 3.0 volts as measured by a bobbin coil. All bobbin indications below 3.0 volts will be allowed to remain in service and all bobbin indications above this limit will be either repaired or removed from service by plugging.

This revision to the voltage repair limits on the hot-leg side reflects a methodology which is significantly different than that contained in GL 95–05. The principal difference between the methodology being applied for the 3.0 volt criteria on the hot-leg side is that the Commonwealth Edison Company (ComEd) is taking credit for the constraint provided by the TSPs to reduce the probability of SG tube burst in the event of a severe accident (i.e., a main steamline break). This constraint is assured by modifying a limited number of SG tubes so that they provide additional stiffness to the TSPs, thereby reducing to a small amount, their deflection under MSLB blowdown loads.

Additionally, inspection and reporting requirements are being added to the Byron 1 and Braidwood 1 TSs in support of the revised voltage-based repair criteria. Further, the maximum permissible value of the iodine-131 concentration in the primary coolant in the Byron 1 TSs is reduced from 1.0 to 0.35 microcuries per gram of coolant. This is the same value for the iodine-131 primary coolant concentration in the Braidwood 1 TSs. Finally, the Bases sections in the Byron 1 and Braidwood 1 TSs are revised to provide a concise description of the methodology proposed by ComEd in support of its proposed revision of the voltage-based SG tube repair criteria.

*Date of issuance:* November 9, 1995.

*Effective date:* November 9, 1995.

*Amendment Nos.:* 77, 77, 69, and 69.

*Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77:* The amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: September 27, 1995 (60 FR 49963).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 9, 1995. The supplemental submittals listed above provide clarifying technical information that does not affect the initial No Significant Hazards Consideration Determination.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* July 5, 1995.

*Brief description of amendment:* This amendment revises Section 6.0 of the Technical Specifications to incorporate several administrative controls and editorial changes to the Training, Plant Review Committee, and Plant Safety and Licensing staff sections.

*Date of issuance:* November 3, 1995.

*Effective date:* November 3, 1995.

*Amendment No.:* 170.

*Facility Operating License No. DPR–20.* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: August 2, 1995 (60 FR 39435).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Van Wylen Library, Hope College, Holland, Michigan 49423.

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

*Date of application for amendments:* April 10, 1995.

*Brief description of amendments:* The amendments revise the required number of operable hydrogen igniters to allow removal of two hydrogen igniters serving the lower reactor cavity and incore instrument cable tunnel.

*Date of issuance:* October 30, 1995.

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

*Amendment Nos.:* 136 and 130.

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

*Date of initial notice in* Federal Register: September 27, 1995 (60 FR 49932).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 30, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

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

*Date of application for amendments:* September 13, 1995.

*Brief description of amendments:* The amendments modify the notation for the overpower delta temperature reactor trip heatup setpoint penalty coefficient as delineated in Note 3 in Technical Specification Table 2.2–1 in order to make the nomenclature consistent with the Standard Technical Specifications and to facilitate a modification to reduce the reactor coolant system hot leg temperature as planned during the Catawba Unit 2 end-of-cycle 7 refueling outage.

*Date of issuance:* October 31, 1995.

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

*Amendment Nos.:* 137 and 131.

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

*Date of initial notice in* Federal Register: September 27, 1995 (60 FR 49933).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 31, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

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

*Date of application for amendments:* September 1, 1995, as supplemented October 17, 1995.

*Brief description of amendments:* The amendments revise Technical Specification (TS) 6.9.1.9 to include references to updated or recently approved methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report. The subject references have been reviewed and approved by the NRC staff.

*Date of issuance:* November 2, 1995.

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

*Amendment Nos.:* 138 and 132.

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

*Date of initial notice in* Federal Register: September 27, 1995 (60 FR 49932). The October 17, 1995, letter provided clarifying information that did not change the scope of the September 1, 1995 application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

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

*Date of application for amendments:* June 13, 1994, as supplemented by letters dated August 15, 1994, March 23, April 18, July 21, and September 22, 1995.

*Brief description of amendments:* The amendments revise the Technical Specifications to increase the initial fuel enrichment limit and establish new loading patterns for new and irradiated fuel in the spent fuel pool to accommodate this increase.

The March 23, 1995, supplement, which provided additional information that modified the June 13, 1994, application's no significant hazards consideration determination, also revises the TS to (1) change the surveillance requirement for boron concentration in the spent fuel pool (SFP), (2) remove the option to use alternate storage configurations in the SFP and replace it with footnotes, (3) add information contained in the Bases to the footnotes, and (4) change the Bases to discuss the option to use specific analyses on alternate fuel.

*Date of issuance:* November 6, 1995.

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

*Amendment Nos.:* 159 and 141.

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

*Date of initial notice in* Federal Register: February 15, 1995 (60 FR 8746); and May 8, 1995 (60 FR 22590). The April 18, July 21, and September 22, 1995, letters provided additional clarifying information that did not change the scope of the June 13, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 1995, and Environmental Assessment dated August 17, 1995.

No significant hazards consideration comments received: No.

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

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

*Date of application for amendment:* May 17, 1995.

*Brief description of amendment:* The amendment will extend the applicability of the current Reactor Coolant System (RCS) Pressure/Temperature Limits and maximum allowed RCS heatup and cooldown rates to 23.6 Effective Full Power Years (EFPY) of operation. In addition, administrative changes were proposed for TS 3.1.2.1 (Boration Systems Flow Paths-Shutdown) and TS 3.1.2.3 (Charging Pump-Shutdown) to clarify the conditions for which a High Pressure Safety Injection pump may be used.

*Date of Issuance:* October 27, 1995.

*Effective Date:* October 27, 1995.

*Amendment No.:* 141.

*Facility Operating License No. DPR–67:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: June 21, 1995 (60 FR 32362).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

*Date of application for amendments:* February 28, 1994.

*Brief description of amendments:* The amendments delete the minimum frequency criteria prescribed for quality assurance audits from Administrative Controls sections 6.5.2.8 and 6.8.4 of the Technical Specifications (TS). Audit periodicity will thereby be controlled by the program described in the Florida Power and Light Company (FPL) Topical Quality Assurance Report.

*Date of Issuance:* October 25, 1995.

*Effective Date:* October 25, 1995.

*Amendment Nos.:* 140 and 80.

*Facility Operating License Nos. DPR–67 and NPF–16:* Amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: April 13, 1994 (59 FR 17599).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 1995.

*No significant hazards consideration comments received*: No.

*Local Public Document Room location:* Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

*Date of application for amendments:* July 26, 1995.

*Brief description of amendments:* These amendments revise selected line items from NRC Generic Letter 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

*Date of issuance:* October 17, 1995.

*Effective date:* October 17, 1995.

*Amendment Nos.:* 177 and 171.

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

*Date of initial notice in* Federal Register: September 13, 1995 (60 FR 47617).

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

*No significant hazards consideration comments received*: No.

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

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

*Date of amendment request:* December 10, 1993.

*Brief description of amendment:* The amendment revises the Cooper Nuclear Station Technical Specifications to change the reporting frequency of the Radioactive Materials Release Report from semiannual to annual and to extend the reporting frequency of the Annual Design Change Report from annual to annually or along with the Updated Safety Analysis Report updates required by 10 CFR 50.71(e). This change reflects revised requirements contained in 10 CFR 50.36a and 10 CFR 50.59(b).

*Date of issuance:* November 3, 1995.

*Effective date:* November 3, 1995.

*Amendment No.:* 172.

*Facility Operating License No. DPR–46*: Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: February 16, 1994 (59 FR 7691).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated Novemver 3, 1995.

*No significant hazards consideration comments received*: No.

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

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

*Date of amendment request:* June 28, 1995.

*Brief description of amendment:* The amendment revises the Cooper Nuclear Station Technical Specifications to increase the required reactor pressure vessel boron concentration, to modify the surveillance frequency for standby liquid control system pump operability testing from monthly to quarterly, and to make editorial changes.

*Date of issuance:* November 8, 1995.

*Effective date:* November 8, 1995.

*Amendment No.:* 173.

*Facility Operating License No. DPR–46:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: August 2, 1995 (60 FR 39441).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 8, 1995.

*No significant hazards consideration comments received*: No.

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

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

*Date of amendment request:* September 5, 1995.

*Description of amendment request:* The amendment modifies the Appendix A Technical Specifications (TSs) for the Turbine Cycle Safety Valves. Specifically, the amendment changes Seabrook Station Appendix A Technical Specification Table 3.7–1 to reduce the Maximum Allowable Power Range Neutron Flux—High Setpoints with Inoperable Main Steam Safety Valves (MSSVs) and Table 3.7–2 to reduce the opening setpoints of the MSSVs. Bases Section 3/4.7.1.1 is changed to include the algorithm used for determining the new setpoint values.

*Date of issuance:* November 2, 1995.

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

*Amendment No.:* 43.

*Facility Operating License No. NPF–86:* Amendment revised the Technical Specifications.

*Date of initial notice in* Federal Register: October 2, 1995 (60 FR 51505).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 1995.

No significant hazards consideration comments received: No.

*Local Public Document Room location:* Exeter Public Library, Founders Park, Exeter, New Hampshire 03833.

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:* December 21, 1994, as supplemented February 22, 1995.

*Brief description of amendment:* The amendment revises the License Condition C.(3), Fire Protection, and certain of the Technical Specifications (TS) related to fire protection requirements. The amendment changes the TS by relocating them to another controlled document, the Technical Requirements Manual referenced in the Final Safety Analysis Report.

*Date of issuance:* November 3, 1995.

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

*Amendment No.:* 191.

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

*Date of initial notice in* Federal Register: February 1, 1995 (60 FR 6303) The February 22, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* August 31, 1995.

*Brief description of amendment:* The amendment revises the Technical Specifications to remove the phrase ''other than Millstone Unit No. 2'' from the Administrative Controls Section 6.3.1, Item (a). This relates to Amendment No. 178 that changed the Technical Specifications to require an individual who serves as the Operations Manager to either hold a Millstone Unit 2 Senior Reactor Operator (SRO) license or have held an SRO license at another pressurized water reactor other than the Millstone Unit No. 2. If the Operations Manager does not hold a Millstone Unit No. 2 SRO license, then an individual serving as the Assistant Operations Manager would be required to possess an SRO license at Millstone Unit 2.

*Date of issuance:* November 2, 1995.

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

*Amendment No.:* 190.

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

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

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

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:* January 27, 1995.

*Brief description of amendments:* The amendments change the Limerick Generating Station Units 1 and 2 Technical Specifications (TS) by eliminating the TS active safety function designation of eight (i.e., four per unit) Drywell Chilled Water System valves.

*Date of issuance:* October 30, 1995.

*Effective date:* October 30, 1995.

*Amendment Nos.:* 103 and 67.

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

*Date of initial notice in* Federal Register: April 26, 1995 (60 FR 20524).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 30, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* November 23, 1994, as supplemented by letter dated August 31, 1995.

*Brief description of amendment:* The proposed changes to the Technical Specifications (TSs) revise TS 4.8.2.1, ''Electrical Power Systems—D.C. Sources,'' Surveillance Requirements, and associated Bases Section 3/4.8.2.

*Date of issuance:* October 31, 1995.

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

*Amendment No.:* 87.

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

*Date of initial notice in* Federal Register: August 2, 1995 (60 FR 39449). The August 31, 1995, letter provided additional and clarifying information that did not change the scope of the November 23, 1994, application and the initial proposed no significant consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 1995.

No significant hazards consideration comments received: No.

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

No significant hazards consideration comments received: No.

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

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

*Date of application for amendment:* November 28, 1994.

*Brief description of amendment:* This amendment revises the technical specifications for the Reactor Coolant System recirculation flow upscale trip function to change the trip setpoint and allowable value to reflect 105% of rated core flow, item one of the above application.

*Date of issuance:* October 31, 1995.

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

*Amendment No.:* 86.

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

*Date of initial notice in* Federal Register: August 2, 1995 (60 FR 39450).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* March 30, 1995, as supplemented August 18, 1995.

*Brief description of amendments:* The amendments eliminate the defined term CONTROLLED LEAKAGE, remove Controlled Leakage flow from the Reactor Coolant System Operational Leakage Limiting Condition for Operation (LCO) and establish a new Seal Injection Flow LCO.

*Date of issuance:* October 30, 1995.

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

*Amendment Nos.:* 178 and 159.

*Facility Operating License Nos. DPR–70 and DPR–75.* The amendments revised the Technical Specifications.

*Date of initial notice in* Federal Register: May 10, 1995 (60 FR 24918). The August 18, 1995, letter provided clarifying information that 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 October 30, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* August 1, 1995, as supplemented by letter dated October 18, 1995.

*Brief description of amendments:* These amendments revise Technical Specification (TS) 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation,'' Table 3.3–3. Table 3.3–3 includes the requirements for the minimum number of toxic gas isolation system (TGIS) trains operable. These amendments are a one-time-only change to extend the allowed TGIS outage times during the replacement of the existing TGIS instrumentation.

*Date of issuance:* November 2, 1995.

*Effective date:* November 2, 1995, to be implemented within 30 days of issuance.

*Amendment Nos.:* Unit 2—Amendment No. 126; Unit 3—Amendment No. 115.

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

*Date of initial notice in* Federal Register: September 13, 1995 (60 FR 47625). The October 18, 1995, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 1995.

No significant hazards consideration comments received: No.

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

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

*Date of application for amendments:* September 30, 1993 (TS–337).

*Brief Description of amendment:* The amendments revise the operating license to reflect issuance of a safety evaluation dated November 2, 1995 accepting the revised Appendix R Safe Shutdown Program to accommodate simultaneous power operation of Browns Ferry Units 2 and 3.

*Date of issuance:* November 2, 1995.

*Effective Date:* November 2, 1995.

*Amendment Nos.:* 226, 241 and 200.

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

*Date of initial notice in* Federal Register: January 5, 1994 (59 FR 629).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 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. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

*Date of application for amendments:* January 4, 1995 (TS 355).

*Brief Description of amendment:* The amendments revise applicability and surveillance requirements for the intermediate power range monitor, average power range monitor (APRM), and APRM Inoperative Trip functions.

*Date of issuance:* November 2, 1995.

*Effective Date:* November 2, 1995.

*Amendment Nos.:* 227, 242 and 201.

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

*Date of initial notice in* Federal Register: June 6, 1995 (60 FR 29888).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 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. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

*Date of application for amendments:* June 2, 1995 (TS 361/371).

*Brief Description of amendment:* The amendments revise the operability definition for residual heat removal service water components for use as a standby coolant supply. The amendments also incorporate related changes to the technical specification Bases which were submitted on October 2, 1995.

*Date of issuance:* November 2, 1995.

*Effective Date:* November 2, 1995.

*Amendment Nos.:* 225, 240 and 199.

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

*Date of initial notice in* Federal Register: August 16, 1995 (60 FR 42610).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 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 No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

*Date of application for amendment:* May 19, 1995; revised September 11, 1995 (TS 95–13).

*Brief description of amendment:* The amendment modifies License Condition 2.C.(17) by extending the required surveillance interval to May 18, 1996, for Surveillance Requirement 4.3.2.1.3 for certain specified engineered safety features response time tests.

*Date of issuance:* October 30, 1995.

*Effective date:* October 30, 1995.

*Amendment No.:* 204.

*Facility Operating License No. DPR–79:* Amendment revises the operating license.

*Date of initial notice in* Federal Register: June 21, 1995 (60 FR 32372); renoticed September 27, 1995 (60 FR 49948).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 1995.

No significant hazards consideration comments received: None.

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

Dated at Rockville, Maryland, this 15th 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.*

[FR Doc. 95–28606 Filed 11–24–95; 8:45 am]

**BILLING CODE 7590–01–P**

## SECURITIES AND EXCHANGE COMMISSION

**[Rel. No. IC–21506; International Series Release No. 886; File No. 812–9704]**

## Banque OBC—Odier Bungener Courvoisier and ABN AMRO Bank N.V.; Notice of Application

November 17, 1995.

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

**ACTION:** Notice of application for exemption under the Investment Company Act of 1940 (the ''Act'').

**APPLICANTS:** Banque OBC—Odier Bungener Courvoisier (''Banque OBC'') and ABN AMRO Bank N.V. (the ''Bank'').

**RELEVANT ACT SECTIONS:** Order requested under section 6(c) of the Act that would exempt applicants from section 17(f) of the Act.

**SUMMARY OF APPLICATION:** Applicants request an order to permit Banque OBC, a subsidiary of the Bank, to act as custodian for investment company assets in The Netherlands.

**FILING DATE:** The application was filed on August 3, 1995 and amended on October 26, 1995.

**HEARING OR NOTIFICATION OF HEARING:** An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on December 12, 1995 and should be accompanied by proof of service on the applicants, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.

**ADDRESSES:** Secretary, SEC, 450 5th Street NW., Washington, D.C. 20549. Applicants, Banque OBC—Odier Bungener Courvoisier, 57 Avenue D'Iena, 75116 Paris, France; ABN AMRO Bank N.V., Foppingadreef 22, 1102 BS Amsterdam, The Netherlands, c/o Edward G. Eisert, Schulte Roth & Zabel, 900 Third Avenue, New York, New York 10022.

**FOR FURTHER INFORMATION CONTACT:** Deepak T. Pai, Staff Attorney, at (202) 942–0574, or Alison E. Baur, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

**SUPPLEMENTARY INFORMATION:** The following is a summary of the application. The complete application may be obtained for a fee from the SEC's Public Reference Branch.

### Applicants' Representations

1. The Bank is a Netherlands banking organization. ABN AMRO Holding N.V. (''Holding'') is the parent company of the Bank, and together with their other domestic and international subsidiaries and affiliates, they constitute the ''ABN AMRO Group.'' As of December 31, 1994, Holding held approximately 100% of the share capital of the Bank, and the Bank accounted for approximately 100% of the total assets of Holding. Both Holding and the Bank are regulated in The Netherlands by De Nederlandsche Bank N.V., the Dutch Central Bank, on behalf of The Netherlands Minister of Finance. At July 31, 1994, Holding ranked 18th in the world, 6th in Europe and 1st in The Netherlands in terms of assets among bank holding companies. At December 31, 1994, Holding had shareholders' equity of approximately U.S. $11.9 billion.

2. Banque OBC, a wholly-owned subsidiary of the Bank, is a French banking institution providing commercial banking, private banking, asset management and merchant banking services to a clientele composed of high net worth individuals, large and medium sized corporations and foreign institutions. Banque OBC is governed by the French Banking Law and is authorized to act, and is monitored by, the Minestere de l'Economie et des Finances, the Banque de France (France's Central Bank) and the Commission Bancaire (France's banking commission). Banque OBC does not meet the minimum shareholders' equity requirement of rule 17f–5.

3. Applicants request an order to permit Banque OBC to maintain custody of securities (''Securities'') of investment companies registered under the Act other than those registered under section 7(d) of the Act (''U.S. Investment Companies''). As used herein, the term ''Securities'' does not include securities issued or guaranteed by the Government of the United States or by any state or any political subdivision thereof, or any agency thereof, or by any entity organized under the laws of the United States or any state thereof (other than certificates of deposit, evidences of indebtedness and other securities, issued or guaranteed by an entity so organized which have been issued and sold outside the United States).

4. Banque OBC would accept deposits of Securities in France only in accordance with a three-party contractual agreement (the ''Agreement''). Each Agreement will be a three-party agreement among (a) the Bank, (b) Banque OBC, and (c) a U.S. Investment Company or its custodian. The Agreement would provide that Banque OBC would provide custodial or sub-custodial services, and the Bank would be liable for any loss to the same extent as if the Bank had been required to provide custody services under such Agreement.

### Applicants' Legal Analysis

1. Section 17(f) of the Act provides that a registered investment company may maintain securities and similar assets in the custody of a bank meeting the requirements of section 26(a) of the Act, a member firm of a national securities exchange, the investment company itself, or a system for the central handling of securities established by a national securities exchange. Section 2(a)(5) of the Act defines ''bank'' to include banking institutions organized under the laws of the United States, member banks of the