services to eligible clients for calendar year 1998. The following service area should not have been included.

State	Service area
North Dakota	MND

Date Issued: June 13, 1997.

Kathleen Welch,

Managing Program Counsel, Office of Program Operations.

[FR Doc. 97–16000 Filed 6–17–97; 8:45 am]

BILLING CODE 7050–01–P

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice (97–087)]

Agency Information Collection: Submission for OMB Review, Comment Request

SUMMARY: The National Aeronautics and Space Administration has submitted to the Office of Management and Budget (OMB) the following proposal for the collection of information under the provisions of the Paperwork Reduction Act (44 U.S.C. Chapter 35).

DATES: Comments on this proposal should be received on or before July 18, 1997.

ADDRESSES: All comments should be addressed to Mr. Richard Kall, Code HK, National Aeronautics and Space Administration, Washington, DC 20546–0001.

FOR FURTHER INFORMATION CONTACT: Ms. Carmela Simonson, NASA Reports Officer, (202) 358–1223.

Title: Small Business and Small Disadvantaged Business Concerns.

OMB Number: 2700–0078.

Type of review: Extension.

Need and Uses: Reports are required to monitor Mentor-Protege performance and progress according to the Mentor Protege Agreement. Reports are internal control to determine if Agency objectives are met.

Affected Public: Business or other for-profit, not-for-profit institutions, State, Local or Tribal Government.

Number of Respondents: 48.

Responses Per Respondent: 2.

Annual Responses: 96.

Hours per Request: 1.

Annual Burden Hours: 96.

Frequency of Report: Semi-annually.

Donald J. Andreotta,

Deputy Chief Information Officer (Operations), Office of the Administrator.

[FR Doc. 97–15957 Filed 6–17–97; 8:45 am]

BILLING CODE 7510–01–M

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice (97–088)]

Agency Information Collection: Submission for OMB Review, Comment Request

SUMMARY: The National Aeronautics and Space Administration has submitted to the Office of Management and Budget (OMB) the following proposal for the collection of information under the provisions of the Paperwork Reduction act (44 U.S.C. Chapter 35).

DATES: Comments on this proposal should be received on or before July 18, 1997.

ADDRESSES: All comments should be addressed to Mr. Richard Kall, Code HK, National Aeronautics and Space Administration, Washington, DC 20546–0001.

FOR FURTHER INFORMATION CONTACT: Ms. Carmela Simonson, NASA Reports Officer, (202) 358–1223.

Title: Uncompensated Overtime.

OMB Number: 2700–0080.

Type of review: Extension.

Need and Uses: For contracts over $500,000, uncompensated overtime information is used to determine (i) whether a contractor will be able to hire and retain qualified individuals, (ii) whether uncompensated overtime hours will be properly accounted, and (iii) the validity of the proposed uncompensated hours.

Affected Public: Business or other for-profit, not-for-profit institutions, State, Local or Tribal Government.

Number of Respondents: 657.

Responses Per Respondent: 1.

Annual Responses: 657.

Hours Per Request: 4.

Annual Burden Hours: 2628.

Frequency of Report: Annually.

Donald J. Andreotta,

Deputy Chief Information Officer (Operations), Office of the Administrator.

[FR Doc. 97–15958 Filed 6–12–97; 8:45 am]

BILLING CODE 7510–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 23, 1997, through June 6, 1997. The last biweekly notice was published on June 4, 1997 (62 FR 30629).

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

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

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

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

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

By July 18, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: May 16, 1997

Description of amendment request: The modification involves replacing the service water (SRW) heat exchangers with new plate and frame heat exchangers having increased thermal performance capability. The saltwater (SW) and SRW piping configuration will be modified as necessary to allow proper fit-up to the new components. A flow control scheme to throttle saltwater flow to the heat exchangers and the associated bypass lines will be added.

Saltwater strainers with an automatic flushing arrangement will be added upstream of each heat exchanger. The majority of the physical work associated with this modification is restricted to the SRW pump room.

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

None of the systems associated with the proposed modification are accident initiators. The SW and SRW Systems are used to mitigate the effects of accidents analyzed in the UFSAR [Updated Final Safety Analysis Report]. The SW and SRW Systems provide cooling to safety-related equipment following an accident. They support accident mitigation functions; therefore, the proposed modification does not increase the probability of an accident previously evaluated.

The proposed modification will increase the heat removal capacity of the SRW System. The design provided under this activity ensures that the safety features provided by the SW and SRW are maintained, and in some instances enhanced; i.e., the availability of important-to-safety equipment required to mitigate the radiological consequences of an accident described in the UFSAR is enhanced by the flexibility and increased thermal margin provided with this design.

The redundant cooling capacity of the SW and SRW Systems have not been altered. Furthermore, the proposed activity will not change, degrade, or prevent actions described or assumed in any accident described in the UFSAR. The proposed activity will not alter any assumptions previously made in evaluating the radiological consequences of any accident described in the UFSAR. Therefore, the consequences of an accident previously evaluated in the UFSAR have not increased.

Therefore, the proposed modification does not involve a significant increase in 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.

The proposed activity involves modifying the SW and SRW System components necessary to support the installation of new SRW heat exchangers. None of the systems associated with this modification are identified as accident initiators in the UFSAR. The SW and SRW Systems are used to mitigate the effects of accidents analyzed in the UFSAR. None of the functions required of the SRW or SW System have been changed by this modification. This activity does not modify any system, structure, or component such that it could become accident initiator, as opposed to its current role as an accident mitigator.

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

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

The safety design basis for the SW and SRW Systems is the availability of sufficient cooling capacity to ensure continued operation of equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with assumptions used in the accident analysis.

The design, procurement, installation, and testing of the equipment associated with the proposed modification are consistent with the applicable codes and standards governing the original systems, structures, and components. The design of instruments and associated cabling ensures that physical and electrical separation of the two subsystems is maintained. Common-mode failure is not introduced by this activity. The equipment is qualified for the service conditions stipulated for that environment. New cable and raceways for this design will be installed in accordance with seismic design requirements. The additional electrical load has been reviewed to ensure the load limits for the vital 1E buses are not exceeded. The circuits and components related to the control valves control loops are safety-related, and are similar to those used for the other safety-related flow control functions. The proposed modification will not have any adverse effects on the safety-related functions of the SW and SRW Systems.

For the above reasons, the existing safety bases have not been altered by the proposed modification. This activity will not reduce the margin of safety as it exists now. In fact, the margin of safety has been increased by this activity due to the increase in the thermal capacity of the dual train design and the increased availability of safety-related components.

Therefore, this proposed modification does not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Alexander W. Dromerick, Acting Director.

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

Date of amendment request: April 23, 1997

Description of amendment request: The proposed changes would revise surveillances 4.3.2.1.1.a, 4.3.2.1.4.b, 4.3.2.1.6.g, 4.3.2.1.10a, 4.3.2.1.10.b, and 4.7.3.b.3 to provide enhanced descriptions of the tests being performed and the tested components.

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:

This change clarification does not involve a significant hazards consideration for the following reasons:

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

The components affected by the proposed changes are not initiators of any accident previously evaluated. The proposed changes to specification 4.3.2.1 items affect only the description of the testing and make no changes in actual operation or testing. The sample heat exchanger valves isolate on receipt of a Safety Injection signal and that feature is unaffected by the additional testing in the proposed change. Therefore, there is no increase in the probability or consequence of a previously analyzed accident.

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

The proposed changes to the surveillance frequencies do not involve physical alterations or additions to plant equipment or alter the manner in which safety-related systems function or are normally operated. The additional testing proposed for the sample heat exchanger valves demonstrates the proper operation of a design feature but does not operate the valve in any new way. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to specification 4.3.2.1 clarify existing testing. The additional testing for the CCW [component cooling water] surge tank level instrumentation adds two components to the surveillance documentation. Therefore, there is no reduction in the margin of safety.

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

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

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

NRC Project Director: Mark Reinhart, Acting.

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

Date of amendment request: April 14, 1997

Description of amendment request: The proposed amendments would revise TS 3/4.3.8, ''Feedwater/Main Turbine Trip System Actuation Instrumentation'' by changing the minimum channels required from 3 to 4. This change reflects a modification that is being installed to correct a design deficiency that could have resulted in a failure to trip the feedwater pumps and main turbine on high water level due to the loss of one of the two instrument lines. The modification adds an auxiliary contact to the trip system logic resulting in an additional channel. The licensee is also proposing to modify the TS action statements for inoperable channels to be similar to TS 3.3.1, ''Reactor Protection System.''

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

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed Technical Specification (TS) change will resolve the common instrument line failure (break) from preventing reactor high water level trip of Feedwater Pumps and Main Turbine. It will not change the probability of occurrence of any accidents, because this instrumentation is not an accident initiator. This instrumentation resolves a potential concern regarding the results of an instrument line break in conjunction with a Feedwater Controller Failure Maximum Demand, which has been postulated and analyzed separately, but are not required to be analyzed in combination, as is described in Chapter 15 of the LaSalle UFSAR. There will not be any increase in probability of feedwater transient (postulated feedwater controller failure with assumed simultaneous failure of one high level trip channel of Feedwater/Main Turbine Trip Actuation Instrumentation), nor an instrument line break. The design change associated with this TS change will prevent the failure of the level 8 trip of Feedwater Pumps and Main Turbine due to loss of common variable water leg of level instrument channels ''B'' and ''C''. Thus there is a slight increase [in] the reliability of the high level trip by assuring that a single instrument failure, including a failure of a sensing line, will not prevent a level 8 trip. The Feedwater/Main Turbine Trip on Reactor Vessel Water Level-High, Level 8, mitigates the consequences of the transient, Feedwater Controller Failure Maximum Demand, due to the main turbine trip with subsequent Turbine Stop Valve closure scram and Reactor Recirculation Pump Trip. This limits the neutron flux peak and fuel thermal transients so that no fuel damage occurs. MCPR remains at or above the operating limit and peak centerline fuel temperature increase is small. The consequences of an accident will not increase, because the redundancy of the instrumentation portion of the Trip Function is somewhat increased.

TS 3.3.8 limiting Condition for Operation (LCO) Actions b and c are proposed to be changed to be similar to the LCO for TS 3.3.1, Reactor Protection System Action b.1 to assure trip capability, while being consistent with the allowed outage times of current TS 3.3.8. Also, the proposed action statements and allowed outage times are consistent with LCO 3.3.2.2, ''Feedwater and Main Turbine High Water Level Trip Instrumentation'', of NUREG 1433, Revision 1, Standard Technical Specifications, General Electric Plants, BWR4, dated April 1995. The limit on continued plant operation of 72 hours in current Action c.1, is overly restrictive, since with one inoperable channel tripped and one Operable channel, the Trip Function is restored to the same status as current Action b.1 (one more instrument failure will cause a failure to actuate on high reactor water level). Therefore, although the proposed Actions are increasing the allowed outage time for the case with only one remaining Operable channel, from 72 hours to 7 days, the level of protection for automatic trip capability is maintained except for a 2 hour period during which trip capability may not exist. In addition, like current Action b.1, the proposed Actions assure that the longest time that automatic trip capability failure due to another instrument failure will exist is 7 days. Therefore, the potential for failure of the Feedwater/Main Turbine trip on reactor vessel high water level may be slightly increased, but is not significant considering the non-safety-related Feedwater Pump and Main Turbine trips are not and are not required to be single-failure proof.

Based on the above, the proposed amendments will not increase the probability or consequences of any accident previously evaluated.

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

The Feedwater/Main Turbine trip is a non-safety function in the non-safety-related feedwater system. The high water level trip is an equipment protective action preventing main steam carry over in the main steam from damaging the main turbine and preventing high pressure liquid discharge through the safety relief valve discharge lines in case of a feedwater transient due to a controller failure to maximum demand. The trip system is not designed to any applicable standards or regulatory guides or 10CFR50 Appendix A General Design Criteria per UFSAR Table 7.1–2. The trip system is not designed nor required to meet the single failure criteria. This is a non-safety/non-divisional trip actuation required in Operating Condition 1, Run Mode, such that high integrity of the trip is maintained. The feedwater system is not required to mitigate the consequences of accidents.

The design change associated with this TS change will increase the reliability of the trip logic. This is accomplished by assuring that a failure of a sensing line will not prevent or cause a level 8 trip. The failure of Feedwater/Main Turbine channel ''C'' trip channel will not have any impact on the RCIC system nor Feedwater/Main Turbine channels ''A'' & ''B'', because the added signal is isolated by a safety-related relay. The 2 out of 3 logic for the trip is maintained.

In addition, the changes to the action statements of the specification do not allow a condition that could cause the actuation instrumentation to fail in a different manner.

Based on the above, the proposed change will not create the possibility of a new or different kind [of accident] from any accident or transient previously evaluated.

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

The proposed TS change will not prevent tripping of Feedwater/Main Turbine or cause false trips. The existing 2 out of 3 logic trip is maintained and does not affect existing failure modes or introduce new failure modes. This change will prevent failure of level 8 trip of Feedwater Pumps and Main Turbine upon loss of common variable water leg for Reactor Vessel Water Level-High, Level 8, instrument channels ''B'' & ''C'' and will slightly increase reliability of the trip logic. Failure of the non-safety-related trip logic will not impact any safety-related system, structure, or component.

The changes to the TS LCO Action statements is consistent with the existing actions, while minimizing the time that automatic trip capability is not maintained. The change from 72 hours allowed operation with one channel Operable and only one channel tripped to 7 days is consistent with the current allowed outage time for only one channel inoperable and not tripped, so any change to the margin of safety provided by the current action requirements is minor.

Based on the above, the proposed TS 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 requested amendments involve no 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.

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

Date of amendment request: May 1, 1997

Description of amendment request: This request changes Technical Specification (TS) Surveillance Requirement (SR) 4.9.A.8.b by clarifying the load value for the emergency diesel generator to be equal to or greater than the largest single load and revise the frequency and voltage requirements during performance of the test.

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

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

The proposed changes represent a clarification of the intent of the performance of the largest single emergency load rejection surveillance for the diesel generator. These changes allow for simulated testing that will more closely duplicate actual emergency loading conditions. By removing the specific load value requirement from the surveillance, the test can be performed using the actual largest load in the same plant configuration that would exist during an actual accident scenario. Verification of the steady-state voltage and frequency within the required time limits provides confidence that the diesel generator can successfully recover from this transient. This provides greater assurance that the diesel generator is capable of performing its intended design function during an accident and the subsequent recovery. The changes to the surveillance requirement will not significantly increase the consequences of an accident previously evaluated.

The diesel generator's design function is to mitigate the consequences of an accident by providing an independent onsite source of alternate AC power with the capacity for operation of systems required to shutdown the reactor and maintain it in a safe shutdown condition until offsite power is restored. The diesel generator and its associated subsystems are not assumed in any safety analysis to initiate any accident sequence for Quad Cities Station; therefore, the probability of an accident previously evaluated is not increased by the proposed amendment.

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

The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Quad Cities Station. The changes revise the largest single emergency load rejection surveillance test acceptance criteria for the diesel generator. This load rejection transient for the diesel generator is bounded by a previously performed accident analysis. This analysis assumes the loss of one diesel generator due to loss of 125 VDC control power for the duration of a LOCA combined with a LOOP. The diesel generator's design function is to mitigate the consequences of an accident by providing an independent onsite source of alternate AC power with the capacity for operation of systems required to shutdown the reactor and maintain it in a safe shutdown condition until offsite power is restored. Only one diesel generator is required to perform this function per unit. Performance of the Surveillance Requirement as proposed provides greater assurance that the diesel generator is capable of performing its intended design function during an accident and the subsequent recovery. No significant changes to existing testing or new modes of facility operation are proposed by this change. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment is required to ensure the diesel generator is tested in accordance with the design basis requirements. The changes represent a revision to the test acceptance criteria for performance of the largest single emergency load rejection surveillance for the diesel generator. This is a possible transient for the diesel generator that is bounded by a previously performed accident analysis. The proposed changes do not adversely affect the capability of the diesel generator to perform its design function. This function is to mitigate the consequences of an accident by providing an independent onsite source of alternate AC power with the capacity for operation of systems required to shutdown the reactor and maintain it in a safe shutdown condition until offsite power is restored. Performance of the Surveillance Requirement as proposed provides greater assurance that the diesel generator is capable of performing its intended design function during an accident and the subsequent recovery. Existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis are not changed. The proposed changes have been evaluated at Quad Cities and found to be acceptable for use based on system design, safety analysis requirements and operational performance. Since the changes maintain the necessary levels of system reliability, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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: May 27, 1997.

Description of amendment request: The proposed amendments would delete from the Technical Specifications (TS) of each unit the specified minimum volume of borated water available to the Standby Makeup Pump; the minimum volume is already specified in other parts of the TS.

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

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

No. This amendment to the Catawba TS maintains the necessary minimum volume of borated water available to mitigate a design basis SSS [standby shutdown system] event through a 72 hour period. Eliminating TS Surveillance 4.7.13.3a.2 does not increase the probability or consequences of any previously evaluated accident, since an adequate borated water source for the SMP [standby makeup pump] is continued to be required by other existing TS.

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

No. This amendment to the Catawba TS continues to ensure that the necessary minimum volume of borated water is available to mitigate an SSS event. The SSS is required to mitigate certain previously evaluated design basis fire, security, and other events. This amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. This amendment changes the TS applicable to an accident mitigating function and does not impact any accident initiator, either new, different, or previously evaluated.

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

No. This amendment continues to ensure that the necessary minimum volume of borated water is available to mitigate an SSS design basis event. The available minimum volume is maintained well above the design basis requirement. Since the source of borated water that is available to supply the SMP continues to be controlled by existing TS (TS 3.7.13.3a.1 and 3.9.10), which both envelope the current 112,320 gallons, sufficient volume has been and will continue to be present to meet design basis requirements. Therefore, no reduction in a margin of safety will result from the changes proposed in this amendment.

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. Paul R. Newton, Legal Department (PB05E), Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242–0001.

NRC Project Director: Herbert N. Berkow.

Duke Power Company, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of amendment request: May 27, 1997

Description of amendment request: The proposed amendment would delete from the Technical Specification of Unit 2 requirements regarding steam generator tube sleeving and repair. These requirements are not applicable to the Westinghouse Model D5 steam generators used by Unit 2.

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

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

No. This amendment to the Catawba Unit 2 Technical Specifications will have no impact on operation of the facility since the change will delete steam generator repair methods that are not applicable to the Catawba Unit 2 steam generators and have not been used to repair the Catawba Unit 2 steam generators.

(2) Will the change create the possibility of a new or different type of accident from any accident previously evaluated?

No. This amendment will delete steam generator repair methods that are not applicable and have not been used. Therefore, the proposed changes will not create the possibility of a new or different accident.

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

No. This amendment will delete steam generator repair methods that are not applicable and have not been used. There will be no impact on safety margins as a result of these changes.

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

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

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Entergy Operations, Inc., et al., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi.

Date of amendment request: May 7, 1997.

Description of amendment request: The amendment request would eliminate selected response time testing (RTT) surveillance requirements (SRs) from the Technical Specifications (TSs) for certain components of the following systems: reactor protection system (SR 3.3.1.1.15), primary containment and drywell isolation instrumentation (SR 3.3.6.1.8), and emergency core cooling system (SRs 3.5.1.8 and 3.5.2.7).

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

1. No significant increase in the probability or consequences of an accident previously evaluated results from this change.

The purpose of the proposed Technical Specification (TS) change is to eliminate response time testing (RTT) requirements for selected components in the Reactor Protection System (RPS), Primary Containment and Drywell Isolation Instrumentation, and Emergency Core Cooling System (ECCS) actuation instrumentation. The Boiling Water Reactor Owners' Group (BWROG) has completed an evaluation which demonstrates that [RTT] is redundant to the other TS-required testing. These other tests, in conjunction with actions taken in response to NRC Bulletin 90–01, ''Loss of Fill-Oil in Transmitters Manufactured by Rosemount,'' and Supplement 1 [to the bulletin], are sufficient to identify failure modes or degradations in instrument response time and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by [RTT] that cannot also be detected by the other TS-required testing. This evaluation was documented in NEDO–32291–A, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' October 1995. EOI [The licensee] has confirmed the applicability of this evaluation to Grand Gulf Nuclear Power Station (GGNS). In addition, EOI will complete the actions identified in the NRC staff's Safety Evaluation of NEDO–32291–A.

Elimination of [ECCS] RTT during MODES 4 and 5 [(i.e., cold shutdown and refueling, respectively)] is acceptable since there are no design basis accidents in MODES 4 and 5 for which the ECCS High Pressure Core Spray (HPCS) system is required to initiate within a specified period of time. The requirement to maintain [ECCS] OPERABLE during Modes 4 and 5 is preserved in the affected Technical Specification. The ECCS RTT required by SR 3.5.1.8 (applicable during MODES 1, 2, and 3, [or power operation, startup, and hot shutdown, respectively]) is adequate to identify any operability problems with the ECCS HPCS system. In addition, during MODES 4 and 5, the probability and consequences of accidents are reduced due to the pressure and temperature limitations of these MODES.

Because of the continued application of other TS-required tests such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response time of these systems [listed in the first paragraph] will be maintained within the acceptance limits assumed in the plant [(GGNS)] safety analyses and required for successful mitigation of an initiating event. The proposed changes do not affect the capability of the associated systems to perform their intended function within their required response time, nor do the proposed changes themselves affect the operation of any equipment.

As a result, EOI has concluded that the proposed changes do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. This change would not create the possibility of a new or different kind of accident from any [accident] previously evaluated.

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

3. This change would not involve a significant reduction in a margin of safety.

The current TS-required response times are based on the minimum allowable values assumed in the plant [(GGNS)] safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed changes do not affect the capability of the associated systems to perform their intended function within the allowable response time used as the basis for the plant safety analyses. The potential failure modes for the components within the scope of this request were evaluated for impact on instrument response time. This evaluation confirmed that, with the exception of loss of fill-oil of Rosemount transmitters, the remaining TS-required testing is sufficient to identify failure modes or degradations in instrument response times and ensure operation of the instrumentation within the scope of this request is within acceptable limits. The actions taken in response to NRC Bulletin 90–01 and Supplement 1 [to the bulletin] are adequate to identify loss of fill-oil failures of Rosemount transmitters. As a result, it has been concluded that plant and system response to an initiating event will remain in compliance with the assumptions of the [GGNS] safety analysis. Elimination of RTT for ECCS HPCS system in MODES 4 and 5 does not reduce the margin of safety since there are no design basis events in MODES 4 and 5 requiring this system to respond in [a] specified period of time from onset of the event. Response time testing required by SR 3.5.1.8 (applicable during MODES 1, 2, and 3) is adequate to identify any equipment or operability concerns).

Further, although not explicitly evaluated, the proposed changes will provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing the potential for inadvertent safety system actuation, reducing plant shutdown risk, limiting radiation exposure to plant personnel [that would be due to the RTT], and eliminating the diversion of key personnel resources to conduct unnecessary testing. Therefore, EOI concluded that this request will result in an overall increase in the margin of safety.

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

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

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 24, 1997.

Description of amendment request: The proposed amendment will modify Technical Specification (TS) 3/4.7.4, Ultimate Heat Sink (UHS), Table 3.7–3, by incorporating more restrictive dry cooling tower (DCT) fan requirements, and it will change the wet cooling tower water consumption in the TS Bases. This proposed amendment seeks to modify the TS to be consistent with revised design basis calculations.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change modifies the UHS TS by not allowing operation with less than 12 DCT fans per DCT. This change is necessary to adequately preserve the assumptions and limits of the revised UHS design basis calculations. These calculations conclude that the UHS is capable of dissipating the maximum peak heat load resulting from the limiting design bases accident (i.e., large break LOCA [large break loss of coolant accident]). The proposed change does not directly affect any material condition of the plant that could directly contribute to causing an accident or that could contribute to the consequences of an accident. The proposed change ensures that the mitigating effects of the UHS will be consistent with the design basis analysis. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

The proposed change modifies the UHS TS to be consistent with revised design basis calculations. The UHS TS is being modified to eliminate operation with less than 12 DCT fans per DCT. The proposed change will not alter the operation of the plant or the manner in which the plant is operated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change modifies the UHS TS by not allowing operation with less than 12 DCT fans per DCT. The proposed change preserves the margin of safety by ensuring that the UHS will be capable of dissipating the maximum design basis accident heat load with adequate margin. Therefore, the proposed 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 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, LA 70122.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 24, 1997

Description of amendment request: The proposed amendment will modify Technical Specifications (TS) 3.1.1.1, 3.1.1.2, 3.10.1 and Figure 3.1–1 by removing the cycle dependent boron concentration and boration flow rate from the Action Statements and removing the ''RWSP at 1720 ppm'' curve from the figure. A change to TS Bases 3/4.1.1.1 and 3/4.1.1.2 has been included to support this change.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The Shutdown Margin requirements are determined by the reload analysis performed every cycle. The Cycle 9 reload analysis has determined that the current Shutdown Margin requirements are acceptable. The proposed change eliminates the reference to 1720 ppm in the Action Statement because 1720 is not adequate to ensure that the Shutdown Margin requirements are met at the beginning of cycle. The proposed Action Statement will continue to ensure that in the event the Shutdown Margin requirements are not met, boration will be immediately initiated to restore the Shutdown Margin to within limits.

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

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

Response: No.

The proposed change does not change the design or configuration of the plant nor does it change how boration systems are operated during normal or accident conditions. It ensures that the Shutdown Margin requirements for accidents already evaluated are promptly restored in the event that the requirements are not met.

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

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

Response: No.

The proposed change has not decreased the amount of Shutdown Margin required. The current Shutdown Margin requirements have been validated by the Reload Analysis for Cycle 9 and are adequate to ensure that the reactor can be made subcritical from all operating conditions, transients, and design basis events. The proposed change ensures that the Shutdown Margin requirements are promptly restored in the event that they are not met. As such, the proposed change ensures that the current margin of safety is maintained.

Therefore, the proposed 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 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, LA 70122.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: June 3, 1997

Description of amendment request: The proposed amendment requests a change to the ACTION Requirements for Technical Specification 3/4.3.2 for the Safety Injection System Sump Recirculation Actuation Signal (RAS). The proposed change will revise the allowed outage time for a channel of RAS to be in the tripped condition from ''prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN'' to the more restrictive time limit of 48 hours and adds a shutdown requirement. Additionally, the 3.0.4 exemption is being removed from the ACTION for the tripped condition. A change to the Technical Specification Basis Section 3/4.3.2 has also been included.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed revision to the TS changes the allowed outage time that a channel of RAS can be in the tripped condition from a maximum of approximately 18 months when one channel is inoperable and 92 days when two channels are inoperable to 48 hours. If a channel were in the tripped condition and a single failure occurred (that of one other channel of RAS), a premature [refueling water storage pool] RWSP low level signal would be generated. During a Design Basis Accident with a containment high pressure condition causing the RWSP outlet check valves to seat, this single failure would prevent the contents of the RWSP from being injected into the reactor coolant system and possibly resulting in failure of both trains of [Emergency Core Cooling System] ECCS and [Containment Spray] CS. Additionally, this would cause the [Low Pressure Safety Injection] LPSI pumps to stop. Reducing the time that a channel of RAS can be placed in the tripped condition will reduce the probability of this scenario occurring during a Design Basis Accident. Since the allowed outage time for a channel of RAS is being limited to 48 hours, this is considered an off-normal operation and a single failure is not required to be postulated during a Design Basis Accident in the accident analysis. Reducing the time the channel can be placed in the tripped condition and thus, the exposure time to this scenario, would not be an accident initiator. The proposed change of being more conservative in the time and condition limits in the TS will not affect the assumptions, design parameters, or results of any accident previously evaluated.

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

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

Response: No.

The proposed change does not change the design or configuration of the plant. The proposed change provides a more conservative allowed outage time for the channel to be in the tripped condition. There has been no physical change to plant systems, structures or components nor will the proposed change reduce the ability of any of the safety-related equipment required to mitigate Anticipated Operational Occurrences or accidents. In fact, this change will potentially increase the ability of safety related equipment to perform its functions. The configuration required by the proposed specification is permitted by the existing specification.

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

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

Response: No.

The proposed change provides a more conservative allowed outage time for the channel to be in the tripped condition. By reducing the allowed outage time, the probability is reduced that a single failure (that of a failure of one channel of RAS with one channel in the tripped condition) would occur that would cause the suction to be prematurely supplied by the Safety Injection System Sump, potentially disabling the [High Pressure Safety Injection] HPSI and CS pumps, and stopping of the LPSI pumps. Therefore, the only change to the margin of safety would be an increase. Since the allowed outage time for a channel of RAS is being limited to 48 hours, this is considered an off-normal operation and a single failure is not required to be postulated during a Design Basis Accident in the accident analysis. The proposed changes do not affect the limiting conditions for operation or their bases.

Therefore, the proposed 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 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, LA 70122.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 9, 1997

Description of amendment request: The proposed amendment would revise the definitions of Limiting Safety System Setting (LSSS) and Instrument/Channel Calibration to reference a new program being added to the Technical Specification (TS) (Section 6.13) for the control of instrument setpoints.

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 amendment will not significantly increase the probability or consequences of any previously-evaluated accidents.

The proposed changes will not result in any direct hardware changes. The change only adds a program to the TS for the establishment and control of instrumentation setpoints that is consistent with current DAEC [Duane Arnold Energy Center] practice. The Instrument Setpoint Control Program is based upon a methodology for the calculation of instrument setpoints that conforms to the guidelines of Regulatory Guide 1.105, Rev. 2. The methodology ensures that adequate margin exists between the normal plant operating conditions and actual instrument setpoints to preclude spurious plant/equipment trips. As a result, the proposed program establishes the criteria for changes in instrument setpoints to ensure that such changes will not result in unnecessary plant transients. Consequently, the probability of any previously-analyzed event is not increased by this change.

The role of the instrumentation and their associated setpoints is in detecting and mitigating plant events and thereby limiting the consequences of any previously-analyzed event. The LSSS[NTSP] and corresponding LTPO[AV] have been developed in accordance with the DAEC Instrument Setpoint Control Program criteria to ensure that the instrumentation remains capable of mitigating events as described in the safety analyses and that the results and consequences described in the safety analyses remain bounding. Therefore, these changes do not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed changes will not create a new or different kind of accident from those previously evaluated.

The proposed changes will not change the method or manner of plant operation, in particular, calibration of TS-required instrumentation. The use of the proposed TS program for the control of changes to instrument setpoints does not impact safe operation of the DAEC in that the design and safety analysis limits will continue to be satisfied. The proposed TS program involves no system additions or physical modifications, other than setpoint changes. Any setpoint changes must conform to the criteria set forth in the TS Instrument Setpoint Control Program. The instrument setpoints are developed using a methodology that conforms to the guidelines contained in Regulatory Guide 1.105, Rev. 2 to ensure the affected instrumentation remains capable of mitigating accidents and transients. Since operational methods remain unchanged and the instrument setpoints have been evaluated to maintain the plant within existing design basis criteria, no new or different type of accident is created.

3. The proposed change will not result in a significant reduction in any margin of safety.

The proposed TS program establishes the DAEC Instrument Setpoint Control Program, which is based upon an NRC-approved methodology. The program establishes the controls and criteria used to establish and revise instrument setpoints. The setpoint calculations use the uncertainties associated with the DAEC instrumentation and actual DAEC physical data and operating practices to ensure the validity of the resulting LTPO[AV] and LSSS[NTSP]. The methodology is based upon combining the uncertainties of the associated channels and takes into account calibration accuracy, instrument uncertainties, drift, etc. The use of this methodology for establishing these setpoints ensures that the design and/or safety analysis limits are not exceeded in any transient or accident. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: May 9, 1997

Description of amendment request: The proposed amendment would revise the definition of Limiting Condition for Operation (LCO) to address the situation when systems, components, etc., are removed from service or otherwise made inoperable during secondary modes of operation, without requiring entry into the LCO actions.

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 amendment will not significantly increase the probability or consequences of any previously evaluated accidents.

The proposed change merely adds criteria to the TS that are consistent with the original design and licensing basis assumptions. Operation in secondary modes of operation (such as surveillance testing, torus cooling mode (test line-up) or Residual Heat Removal system, and use of High Pressure Coolant Injection system or Reactor Core Isolation Cooling system in test line-up for reactor pressure control during transients) is assumed in the safety analysis report (Ref. UFSAR Section 6.3.4.2.1 and 7.3.4.2). Because no changes in actual equipment operation or testing are being made as part of this change, the probability of any event which could be induced by such operation or testing is not increased. Also, the change will ensure that the time such equipment is removed from service is kept very short in duration, either through existing TS Allowed Outage Time (AOT) notes or administratively by procedures. This is consistent with the assumption that the time in such secondary modes of operation (*i.e.,* safe test interval) is much shorter than the allowable repair time (*i.e.,* LCO time). Therefore, the proposed change will not significantly increase the probability of any previously evaluated accident.

The uniform application of the new TS criteria will further ensure that the plant remains within the original design and licensing basis assumptions for equipment removed from service during secondary modes of operation. In particular, in the special case where testing also removes the redundant system, train, component, etc., from service, these criteria ensure that both affected systems, trains, etc., are properly controlled. This is acceptable because the time in such secondary modes of operation is very short in duration, such that the impact on the overall availability/reliability is insignificant. Therefore, the consequences of any previously analyzed accident are not significantly increased by this change.

2. The proposed changes will not create a new or different kind of accident from those previously evaluated.

The proposed changes will not add a new or different kind of accident because the plant will not be operated in a different way. Operation in secondary modes has been previously evaluated and found to be acceptable (Ref. General Electric reports APED–5736: *Guideline for Determining Safe Test Intervals and Repair Times for Engineered Safeguards, and NEDO–10739: Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems*). The proposed change merely adds criteria to the TS that are consistent with the assumptions contained within these evaluations. Consequently, no new or different accidents are postulated as a result of this proposed change.

3. The proposed change will not result in a significant reduction in any margin of safety.

Because the criteria being added to the TS enforce the assumptions of the evaluations that form the basis of the existing TS (Ref. TS Bases 4.1, 4.2, and 3.5), the proposed change will not result in a significant reduction in any margin of safety.

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

Local Public Document Room location: Cedar Rapids Public Library, 00 First Street, SE., Cedar Rapids, Iowa 52401.

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment requests: December 20, 1996

Description of amendment requests: The proposed amendments would reduce the frequency and scope of reactor coolant pump flywheel inspections.

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

We have evaluated the proposed T/S changes and have determined they do not represent a significant hazards consideration based on the criteria established in 10 CFR 50.92(c). Operation of Cook Nuclear Plant in accordance with the proposed amendment will not:

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

This change will reduce the frequency and scope of the surveillance testing on the reactor coolant pump flywheels. Operating power plants have been inspecting their flywheels for over 20 years with no flaws identified which affect flywheel integrity. Past examinations performed to satisfy T/S 4.4.10.1 have not revealed any cracking of flywheel plates at Cook Nuclear Plant. Crack extension over a 60 year service life is negligible. Structural reliability studies have shown that eliminating inspections after 10 years of plant life will not significantly change the probability of failure. Most flaws which could lead to failure would be detected during preservice inspection or, at worst, early in plant life, and crack growth over plant life is negligible. As stated in the SER associated with WCAP–14535, assuming an initial crack of 10% of the distance from the keyway to the flywheel outer radius and a maximum fatigue crack growth, ASME margins would be maintained during the 10-year inspection period. Therefore, the change in test frequency will not endanger public health or safety. For these reasons, it is our belief the proposed changes do not involve a significant increase in the probability or consequences of a previously evaluated accident.

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

The changes will not introduce any new modes of plant operation, nor will any physical changes to the plant be required. Thus, the changes will not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

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

This change will reduce the frequency and scope of the surveillance testing on the reactor coolant pump flywheels. Operating power plants have been inspecting their flywheels for over 20 years with no flaws identified which affect flywheel integrity. Past examinations performed to satisfy T/S 4.4.10.1 have not revealed any cracking of flywheel plates at Cook Nuclear Plant. Crack extension over a 60 year service life is negligible. Structural reliability studies have shown that eliminating inspections after 10 years of plant life will not significantly change the probability of failure. Most flaws which could lead to failure would be detected during preservice inspection or at worst early in plant life, and crack growth over plant life is negligible. As stated in the SER associated with WCAP–14535, assuming an initial crack of 10% of the distance from the keyway to the flywheel outer radius and a maximum fatigue crack growth, ASME margins would be maintained during the 10-year inspection period. For these reasons, it is our belief the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: April 30, 1997

Description of amendment request: The proposed amendment would remove Technical Specifications (TSs) regarding meteorological monitoring instrumentation in accordance with NRC Generic Letter (GL) 95–10, ''Relocation of Selected Technical Specification Requirements Related to Instrumentation.'' Specifically, the amendment would delete TS 3/4.3.7.3, ''Meteorological Monitoring Instrumentation,'' including associated TS Tables 3/4.3.7.3–1, and TS Bases 3/4.3.7.3. The TS Index would be revised to show these deletions. The deletion of TS 3.3.7.3 would also eliminate the requirement that a Special Report to be submitted to the NRC pursuant to TS 6.9.2 when one or more meteorological monitoring instrumentation channels is inoperable for more than 7 days. The licensee states that the deleted requirements would be relocated to the Updated Safety Analysis Report (USAR), except that the special reporting requirement would be discontinued as the licensee would continue to evaluate future inoperability of meteorological instrumentation for reportability in accordance with 10 CFR 50.72 and 10 CFR 50.73. The licensee will also insert the word ''nominal'' in the relocated tables in the USAR to indicate that the meteorological instrumentation elevations of 30 and 200 feet are nominal elevations (this change would be made because, as the licensee reported in LER 96–14, the actual locations of the air temperature monitoring instruments are 26.8 feet and 194.8 feet and the actual locations of the wind indicator (speed and direction) monitoring instruments are 30.9 feet and 199.4 feet). As stated in GL 95–10, the NRC staff has determined that meteorological monitoring instrumentation does not serve such a primary protective function as to warrant inclusion in the TS in accordance with 10 CFR 50.36 criteria. Thus, in GL 95–10, the NRC staff established that relocation of the meteorological instrumentation requirements to the USAR (whereby changes are controlled by the licensee pursuant to 10 CFR 50.59) is 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:

1. The operation of Nine Mile Point Unit 2 [NMP2], in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The NMP2 meteorological monitoring instrumentation is used to provide data for use in radioactive dose assessment with respect to routine or accidental releases of radioactive materials to the atmosphere. The deletion of the special reporting requirements is an administrative change. The subject special reporting requirements serve no nuclear related protective function. The relocation of the meteorological monitoring instrumentation requirements from the TSs to the USAR, and the addition of the word nominal to the USAR and tables, will not increase the probability of an accident since the specification applies only to monitoring instrumentation. This also is an administrative change and does not reduce the effectiveness of the current instrumentation requirements. The meteorological monitoring instrumentation requirements are not precursors to any accident previously evaluated. According to the NRC Staff (GL 95–10), the meteorological monitoring instrumentation does not serve to ensure the plant is operated within the bounds of initial conditions assumed in any design basis accidents or transients previously evaluated, or that the plant will be operated to preclude transients or accidents. In addition, the meteorological monitoring instrumentation does not function as part of the primary success path of a safety sequence analysis used to demonstrate that the consequences of these events are within the appropriate acceptance criteria. Therefore, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed deletion of the special reporting requirements is an administrative change. The subject special reporting requirements serve no nuclear related protective function. The proposed change also removes meteorological monitoring instrumentation specifications from the NMP2 TSs. This also is an administrative change and does not reduce the effectiveness of the current instrumentation requirements. The relocation of the meteorological instrumentation requirements to the USAR, and the addition of the word nominal to the USAR and tables, will not create the possibility of a new or different kind of accident since the specification only applies to monitoring instrumentation. The NRC Staff has concluded in GL 95–10 that the provisions of the meteorological monitoring instrumentation specifications are not related to dominant contributors to plant risk. The NMP2 meteorological instrumentation is used to provide data for use in radioactive dose assessment with respect to routine or accidental releases of radioactive materials to the atmosphere. Since no physical modification to the plant is being performed, and no changes to actual plant operations are required by the change, removal of the specifications from the NMP2 TSs will not create the possibility of a new or different kind of accident from any previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed deletion of the special reporting requirements is an administrative change. The subject special reporting requirements serve no nuclear related protective function. The proposed removal of the instrumentation requirements from the NMP2 TSs is also an administrative change and does not reduce the effectiveness of the current instrumentation requirements. The relocation of the meteorological instrumentation requirements to the USAR, and the addition of the word nominal to the USAR and tables, will not involve a reduction in a margin of safety since the specification only applies to monitoring instrumentation. The instrumentation will continue to meet the requirements of Regulatory Guide 1.23, and the offsite dose calculations will continue to use the actual measured elevation differences. In GL 95–10, the NRC Staff concluded (1) That the meteorological monitoring instrumentation does not function as part of the primary success path of a safety sequence analysis, and (2) that the meteorological monitoring instrumentation specifications are not related to dominant contributors to plant risk. Therefore, the removal of the meteorological monitoring instrumentation specifications from the NMP2 TSs will not result in a significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: Alexander W. Dromerick, Acting Director.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: May 15, 1997

Description of amendment request: The proposed amendment would revise Technical Specification Sections 3.1 and 4.1 ''Reactor Protection System'' and the associated Bases to remove run mode intermediate range monitor high flux/inoperative with the associated average power range monitor downscale scram trip function and incorporate editorial revisions.

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

1. The operation of Millstone Nuclear Power Station, Unit No. 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

No physical change is being made to any systems or components that are credited in the safety analysis, therefore there is no change in the probability or consequences of any accident analyzed in the UFSAR [Updated Final Safety Analysis Report].

The design basis accident applicable to the startup power region is the Control Rod Drop Accident (CRDA). The UFSAR does not credit the RUN Mode IRM [intermediate range monitor] High Flux/Inoperative with the associated APRM [average power range monitor] downscale scram Trip Function (IRM RUN Mode SCRAM) in the termination of this accident. Accident mitigation is provided by the APRM 120% power scram. Therefore, elimination of the IRM RUN Mode SCRAM function has no adverse affect on previously evaluated accidents.

The Continuous Control Rod Withdrawal Error (CWE) transient is terminated by the Rod Block Monitor (RBM) in the RUN Mode. The APRM Reduced High Flux Scram provides the primary STARTUP Mode protection in conjunction with the IRMs and limits the consequences of this transient. Therefore, elimination of the IRM RUN Mode SCRAM function has no effect on the consequences of this transient.

Clarification of the LCO [limiting condition for operation] RPS [reactor protection system] Table aligns requirements with Limiting Safety System Settings. Further revisions to LCO 3.1 Reactor Protection System Table 3.1.1 and associated TS [technical specification] bases to clarify APRM Trip Functions do not alter the required trip functions. Deletion of RUN requirement and associated Action B for Reduced High Flux fixes an editorial error introduced in a previous amendment. This trip function is not effective with the mode switch in the RUN position and removal does not alter the neutron monitoring requirements credited in the accident analyses.

Adding a new surveillance to verify SRM [source range monitor]/IRM/APRM overlap will enhance neutron monitoring during startups and shutdowns and does not have an adverse affect on previously evaluated accidents.

None of the proposed changes will affect any of the rod blocks or other precursor events to either the CRDA or CWE. Therefore, there is no change in the probability of any accident previously analyzed.

2. The operation of Millstone Nuclear Power Station, Unit No. 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes affect only the operations of neutron monitoring and protective systems (IRM and APRM) which provide indication and mitigation actions only. Operation of these systems does not create the possibility for new precursors (such as reactivity) which would introduce a new or different kind of accident from any accident previously evaluated.

Additionally, the proposed changes do not affect the ability of those systems required to mitigate previously evaluated accidents during the modes they are credited.

3. The operation of Millstone Nuclear Power Station, Unit No. 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The only scram function that the UFSAR takes credit for in the mitigation of the limiting accident (control rod drop accident) is the APRM 120% power scram which is not affected by this change. Only the IRM RUN Mode SCRAM, for which the UFSAR takes no credit in the termination of any analyzed event, is removed by this change. Removal of the IRM RUN Mode SCRAM will avoid the need to operate the plant in a ''half scram'' condition with the potential for an inadvertent plant transient. For these reasons, the change does not involve a significant reduction in a margin of safety.

The Continuous Control Rod Withdrawal Error (CWE) transient is terminated by the Rod Block Monitor (RBM) in the RUN Mode. When initiated from the STARTUP Mode, the consequences of a CWE are limited by the APRM Reduced High Flux scram in conjunction with the IRM scram function. Therefore eliminating the TS requirement for the IRM RUN Mode SCRAM will not reduce the margin of safety for this transient.

Clarification of the LCO RPS Table aligns requirements with Limiting Safety System Settings. Further revisions to LCO 3.1 Reactor Protection System Table 3.1.1 and associated TS bases to clarify APRM Trip Functions do not alter the required trip functions. Deletion of the RUN requirement and associated Action B for Reduced High Flux corrects an editorial error introduced in a previous amendment. This trip function is not effective with the mode switch in the RUN position and removal does not alter the neutron monitoring requirements credited in the accident analyses.

Adding a new surveillance to verify SRM/IRM/APRM overlap will enhance neutron monitoring during startups and shutdowns and consequently does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community—Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: May 20, 1997

Description of amendment request: This submittal supersedes the January 22, 1996, submittal which was previously noticed on February 28, 1996 (61 FR 7554). The proposed change would relocate the containment isolation valve (CIV) list, Table 3.6–2, from the Technical Specifications to the Technical Requirements Manual (TRM). This change would affect Technical Specification Sections 1.8.1.b, 4.6.1.1.a, 3.6.3.1, 4.6.3.1.1, and 4.6.3.1.2, and Basis Section 3/4.6.3. A note at the bottom of Table 3.6–2 regarding the CIVs that are subject to administrative controls is retained in the Technical Specifications by relocating it to Sections 1.8.1.b and 3.6.3.1. This change is being performed in accordance with Generic Letter 91–08, which provides guidance for removal of component lists from the Technical Specifications.

Additionally, a change to provide relief in the surveillance requirement in Section 4.6.1.1.a is included. The change allows valves, blind flanges, and deactivated automatic valves located inside the containment and are locked, sealed, or otherwise secured in the closed position to be verified closed prior to entering Mode 4 from Mode 5, if not performed within the previous 92 days. The current requirements check the valve position once per 31 days.

TS Bases Section 3/4.6.3 is updated to reflect the removal and relocation of the CIV list to the TRM. Also, details of the administrative controls for operating CIVs while in Modes 1 through 4 are added to Bases Section 3/4.6.3.

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

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

The proposed change to relocate the containment isolation valve (CIV) list will not result in any hardware or equipment operating changes. The proposed change is based on Generic Letter (GL) 91–08 and merely relocates the CIV table and removes all references to the table. The relocation of the CIV table from the Technical Specifications does not affect the operability requirements of any of the listed valves. Technical Specifications will still continue to require the CIVs to be operable. The LCO [limiting condition for operation] and surveillance requirements for the valves will remain in Technical Specifications. The CIV table will be relocated to the Millstone Unit No. 2 Technical Requirements Manual (TRM), which is controlled in accordance with 10 CFR 50.59. This change does not alter the design, function, or operation of the valves involved. Thus, there is no significant affect on the possibility or consequences of any previously evaluated accident.

The change to Surveillance Requirement (SR) 4.6.1.1.a will allow the valves, blind flanges and deactivated automatic valves located inside the containment that are locked, sealed, or otherwise secured in the closed position to be verified closed prior to entering Mode 4 from Mode 5, if not performed within the previous 92 days, instead of the current 31 day requirement. This means that the surveillance interval could be as long as the entire operating cycle, depending on whether entry into Mode 5 is required during the cycle. The change in the surveillance frequency (increase in time from 31 days to not less than 92 days and only prior to entering Mode 4 from Mode 5) recognizes that these valves are operated under administrative controls and probability of misalignment is low. This provides adequate assurance that the containment function assumed in the accident analysis will be maintained. Therefore, there is no significant affect on the probability or consequences of any previously evaluated accident. This proposed change to SR 4.6.1.1.a is consistent with NUREG–1432 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors Revision 1 (SR 3.6.3.4).

The information added to the Bases will provide additional guidance to ensure the plant is operated correctly. This information will not result in any new approaches to plant operation. Therefore, there is not significant affect on the probability or consequences of any previously evaluated accident.

These proposed changes do not alter the design, function, or operation of the valves involved. Therefore, there is no significant increase in the probability or consequence of an accident previously evaluated.

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

The change to relocate the CIV list from the Technical Specifications to the TRM will not impose any different operational or surveillance requirements, nor will the change remove any such requirements. Adequate control will be maintained. Furthermore, as stated above, the proposed change does not alter the design, function, or operation of the valves involved, and therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change to SR 4.6.1.1.a reduces the surveillance frequency for valves, blind flanges and deactivated automatic valves located inside the containment. It does not alter the design, function, or operation of the valves. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The information added to the Bases will provide additional guidance to ensure the plant is operated correctly. This information does not alter the design, function, or operation of the valves involved. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The effectiveness of Technical Specifications will be maintained since the change will not alter function or operability requirements for any CIV. In addition, the relocation of the valve list is consistent with the guidance provided in GL 91–08, and the change to the surveillance interval is consistent with NUREG–0212 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors Revision 2 (LCO 3.6.1.1) and NUREG–1432 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors Revision 1 (LCO 3.6.3).

The information added to the Bases is consistent with the guidance provided in GL 91–08 and NUREG–1432 Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors Revision 1. The intent of the Technical Specifications will be met since this information will not result in any new approaches to plant operation.

Therefore, there is no significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: May 9, 1997

Description of amendment request: The proposed amendment would revise the shutdown margin requirements and add Technical Specification 3/4.3.5 to provide the limiting condition for operation (LCO) and surveillance requirements for the shutdown margin monitors. The proposed amendment would also make administrative changes and revise the associated Bases section.

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:

NNECO has reviewed the proposed changes in accordance with 10 CFR 50.92 and has concluded that the change does not involve a significant hazards consideration (SHC). The bases for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed changes do not involve [an] SHC because the changes would not:

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

The proposed Technical Specification changes will revise the current shutdown margin requirements for Modes 3, 4 and 5 in Figures 3.1–1, 3.1–2, 3.1–3, 3.1–4 and 3.1–5 and allow for additional boration of the RCS [reactor coolant system] as directed by Specification 3.3.5. The new Shutdown Margin requirements are based on re-analyses of the Boron Dilution Event provided by Westinghouse. In the re-analyses, assumptions were modified in order to justify the operability of the Shutdown Margin Monitor for count rates which are lower than currently allowed. The proposed Shutdown Margin requirements for Modes 3, 4 and 5 will continue to assure that the operator has a minimum of 15 minutes from the alarm to loss of shutdown margin during an assumed Boron Dilution Event.

The proposed change also adds Technical Specification 3/4.3.5 to provide the LCO and Surveillance Requirements for the Shutdown Margin Monitors. LCO 3.3.5 refers to the Core Operating Limits Report (COLR) which will specify the minimum count rate/alarm ratio requirements in order to consider the Shutdown Margin Monitors operable. The LCO also directs the additional boration of the RCS in order to allow the Shutdown Margin Monitors to be considered operable for lower count rates. Also, a footnote (**) is included in Specification 3/4.3.5 to make the Specification treatment of the valves consistent with the Mode 6 and Mode 5-loops drained requirements.

Due to the addition of Technical Specification 3/4.3.5, the related Bases information is added as BASES Section 3/4.3.5. Additionally, the Bases information for the Shutdown Margin Monitors which is currently in BASES Section 3/4.3.1 is moved to the added BASES Section 3/4.3.5. This Bases information is also revised to be consistent with the added Specification 3/4.3.5.

Also, due to the addition of Technical Specification 3/4.3.5, the guidance related to the Shutdown Margin Monitor in Tables 3.3–1 and 4.3–1 is deleted to avoid redundancy.

Additionally, Section 3/4.1.2 of the Bases is revised so that it refers to Figure 3.1–4 (Shutdown Margin for Mode 5/filled) instead of Figure 3.1–5 (Shutdown Margin for Mode 5/drained). This change will make the Bases consistent with the ACTION statement requirements of Technical Specifications 3.1.2.2 and 3.1.2.6.

Finally, Reference 12 (NUSCO–152, Addendum 4) is added to the list of references in Section 6.9.1.6.b. The addition of this reference is considered administrative and is not related to or required by the changes proposed for the Shutdown Margin requirements or Shutdown Margin Monitors.

The new requirements for increased Shutdown Margin (Figures 3.1–1 to 3.1–5) and additional boration (LCO 3.3.5) continue to assure that the operator will have a response time of at least 15 minutes to mitigate the consequences of a Boron Dilution Event. The implementation of the new requirements does not alter the alignment of any plant equipment and therefore, the change cannot increase the probability or consequences of any previously analyzed accident.

The proposed changes will not adversely affect the assumptions or results of other FSAR [Final Safety Analysis Report] accident analysis and it is concluded that this change is safe. The changes do not adversely affect any equipment credited in the safety analysis.

Based upon the re-analyses of the boron dilution event, revised plant operating requirements (shutdown margin) are generated to maintain the required operator action time. Therefore, there is no effect on the probability of occurrence or consequences of previously evaluated accidents.

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

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

The proposed Shutdown Margin requirements for Modes 3, 4 and 5 (Figures 3.1–1 to 3.1–5 and additional boration as per Specification 3.3.5) will continue to assure that the operator has a minimum of 15 minutes from the alarm to loss of shutdown margin during an assumed Boron Dilution Event. Additionally, the use of these revised requirements allows the Shutdown Margin Monitor to be considered operable for count rates which are lower than currently allowed.

The changes do not introduce any new failure modes or malfunctions since the changes implement revised, more conservative plant operating requirements (shutdown margin) which are based on re-analyses of the Boron Dilution Event. Also, the changes do not eliminate any existing requirements.

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

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

The proposed Shutdown Margin requirements for Modes 3, 4 and 5 (Figures 3.1–1 to 3.1–5 and additional boration as per Specification 3.3.5) will continue to assure that the operator has a minimum of 15 minutes from the alarm to loss of shutdown margin during an assumed Boron Dilution Event. Additionally, the use of these revised requirements allows the Shutdown Margin Monitor to be considered operable for count rates...which are lower than currently allowed.

The re-analyses of the Boron Dilution Event demonstrated that the required operator action time is maintained. As such, the re-analyses will become the ''analysis of record'' for the Boron Dilution Event in Modes 3, 4 and 5. The Boron Dilution Event analysis is documented in FSAR Chapter 15.4.6.

The re-analyses of the Boron Dilution Event and the proposed revisions to the Technical Specifications do not adversely affect the results of the current FSAR accident analysis and therefore, it is concluded that this change is safe. Additionally, the change does not adversely affect any equipment credited in the safety analysis.

The changes do not have an adverse impact on the protective boundaries and there is no reduction in the margin of safety as specified in the Technical Specifications. Thus, this proposed change does not involve a significant reduction in the margin of safety.

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

In conclusion, based on the information provided, it is determined that the proposed changes do not involve an SHC.

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

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: May 14, 1997

Description of amendment request: Technical Specification Surveillance Requirement 4.8.2.1.c.4 requires that each battery charger be tested to verify that it can supply a specified current at 125 volts. The proposed amendment would increase the required test voltage.

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:

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

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

The proposed changes to Technical Specification Surveillance 4.8.2.1.c.4 to increase the required test voltage for the battery chargers from 125 volts to greater than or equal to 132 volts is consistent with the design criteria of the chargers and performing battery charger surveillance testing does not significantly increase the probability of an accident previously evaluated. The proposed changes to increase the required test voltage for the battery chargers provides the necessary assurance that the battery chargers will function as required in previous evaluations and does not significantly increase the consequence of an accident previously evaluated.

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

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

The proposed changes to Technical Specification Surveillance 4.8.2.1.c.4 to increase the required test voltage for the battery chargers from 125 volts to greater than or equal to 132 volts does not change the operation of the battery chargers during normal or accident evaluations.

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

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

The proposed change to Technical Specification Surveillance 4.8.2.1.c.4 to increase the required test voltage for the battery chargers from 125 volts to greater than or equal to 132 volts provides assurance that the battery chargers are capable of supplying the largest combined demands of the various steady state loads, plus the current required to recharge its battery, which has undergone a duty cycle discharge, to its fully charged condition in less than 24 hours.

Therefore, the proposed revision does not involve a significant reduction in a margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: March 3, 1997 as supplemented by letter dated May 5, 1997. The May 5, 1997, supplement revised the proposed no significant hazards consideration entirely

Description of amendment request: The proposed changes to the Hope Creek (HC) Technical Specifications (TSs) would: (1) Change TS 3/4.3.1, ''Reactor Protection System Instrumentation,'' TS 3/4.3.2, ''Isolation Actuation Instrumentation,'' and TS 3/4.3.3, ''Emergency Core Cooling System Actuation Instrumentation'' to include additional information concerning response time testing; (2) Change TS 4.0.5 to reference inservice inspection and test requirements; (3) Change TS 3/4.6.1, ''Primary Containment,'' and associated Bases to reflect a design modification; (4) Change TS 3/4.7.7, ''Main Turbine Bypass System,'' to specify a new operability requirement; and (5) Change the Bases for TS 3/4.8, ''Electrical Power Systems.''

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

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

The proposed changes for the TS related to response time testing reflect testing methodologies that were approved by the NRC in Amendment No. 85 to the Hope Creek TS. These proposed TS revisions involve: (1) no hardware changes; (2) no significant changes to the operation of any systems or components in normal or accident operating conditions; and (3) no changes to existing structures, systems or components. Therefore, these changes will not increase the probability of an accident previously evaluated. Since the plant systems associated with these proposed changes will still be capable of: (1) meeting all applicable design basis requirements; and (2) retain the capability to mitigate the consequences of accidents described in the HC [Updated Final Safety Analysis Report] UFSAR, the proposed changes were determined to be justified. As a result, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to Surveillance Requirement 4.0.5 do not alter the current requirements for the Hope Creek inservice inspection and inservice testing programs and are considered to be editorial in nature. These proposed TS revisions involve: (1) no hardware changes; (2) no significant changes to the operation of any systems or components in normal or accident operating conditions; and (3) no changes to existing structures, systems or components. Therefore, these changes will not increase the probability of an accident previously evaluated. Since the plant systems associated with these proposed changes will still be capable of: (1) Meeting all applicable design basis requirements; and (2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to the drywell and suppression chamber purge system are being made to justify design modifications to that system. As discussed in NRC Notice of Violation 50–354/96–10–01, this design modification replaced isolation valves containing resilient material seals with metal seated valves under 10CFR50.59. As a result of this modification, a 24 month frequency has been implemented to perform Type C tests on these new metal seated valves. PSE&G has concluded that the 24 month frequency is appropriate for the new valves since: (1) This frequency is imposed by Surveillance Requirement 4.6.1.2.d, which is applicable to similar containment isolation valves in Table 3.6.3–1 that penetrate the primary containment; and (2) concerns raised about severe environment-induced degradation and frequent use for the previously installed resilient seal material valves are not applicable to the replacement metal seat valves. PSE&G has concluded that the valve modification was an enhancement to the Hope Creek design that did not impact the isolation capability of the drywell and suppression chamber purge system. No significant changes were made to the operation of these valves in normal or accident operating conditions. As a result, these changes will not increase the probability of an accident previously evaluated. Since the plant systems associated with these proposed changes will still be capable of: (1) Meeting all applicable design basis requirements; and (2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to [Limiting Condition for Operation] LCO 3.7.7 establish consistent and appropriate requirements for main turbine bypass valve operability requirements. These changes do not impact the assumptions contained in these UFSAR analyses since they do not change the manner in which Hope Creek is currently permitted to operate. Since the ACTION Statement for LCO 3.7.7 already allows indefinite continued operation below 25% of RATED THERMAL POWER with an inoperable main turbine bypass valve system, the proposed modification to the APPLICABILITY statement for this LCO does not involve: (1) Hardware changes; (2) significant changes to the operation of any systems or components in normal or accident operating conditions; or (3) changes to existing structures, systems or components. Therefore these changes will not increase the probability of an accident previously evaluated. Since the plant systems associated with these proposed changes will still be capable of: (1) meeting all applicable design basis requirements; and (2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

The proposed changes to the HC emergency diesel generator (EDG) TS Bases [Change 5—Bases for TS 3/4.8, ''Electrical Power Systems''] include information contained in the Safety Evaluation Report for Technical Specification Amendment No. 75. This information concerns the bases for the allowed-outage-time (AOT) for the C and D EDGs. Concerning the revisions to planned C and D EDG outages, PSE&G believes that implementation of 10CFR50.65 requirements to monitor EDG unavailability will provide an acceptable and more clearly defined method for maintaining EDG availability within acceptable limits. As stated in PSE&G's letter LR-N97167, dated March 21, 1997, Hope Creek will not plan C or D EDG outages that exceed 72 hours if the total unavailability of the EDG will be greater than 720 hours on a 12 month rolling basis. The proposed TS revisions involve: (1) no hardware changes; (2) no significant changes to the operation of any systems or components in normal or accident operating conditions; and (3) no changes to existing structures, systems or components. Therefore these changes will not increase the probability of an accident previously evaluated. Since the plant systems associated with these proposed changes will still be capable of: (1) Meeting all applicable design basis requirements; and (2) retain the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes for the TS related to response time testing reflect testing methodologies that were approved by the NRC in Amendment No. 85 to the Hope Creek TS and are being made to clarify the licensing basis for performing response time testing. The proposed changes will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: (1) No hardware changes; (2) no significant changes to the operation of any systems or components; and (3) no changes to existing structures, systems or components, there can be no impact on the occurrence of an accident previously evaluated. Furthermore, there is no change in plant testing proposed in this change request that could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to Surveillance Requirement 4.0.5 do not alter the current requirements for the Hope Creek inservice inspection and inservice testing programs and are considered to be editorial in nature. The proposed changes will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: (1) No hardware changes; (2) no changes to the operation of any systems or components; and (3) no changes to existing structures, systems or components, there can be no impact on the occurrence of an accident previously evaluated. Furthermore, there is no change in plant testing proposed in this change request that could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the drywell and suppression chamber purge system are being made to justify design modifications to that system. As discussed in NRC Notice of Violation 50–354/96–10–01, this design modification replaced isolation valves containing resilient material seals with metal seated valves under 10 CFR 50.59. As a result of this modification, a 24 month frequency has been implemented to perform Type C tests on these new metal seated valves. PSE&G has concluded that the 24 month frequency is appropriate for the new valves since: (1) This frequency is imposed by Surveillance Requirement 4.6.1.2.d, which is applicable to similar containment isolation valves in Table 3.6.3–1 that penetrate the primary containment; and (2) concerns raised about severe environment-induced degradation and frequent use for the previously installed resilient seal material valves are not applicable to the replacement metal seat valves. PSE&G has concluded that the valve modification was an enhancement to the Hope Creek design that did not impact the isolation capability of the drywell and suppression chamber purge system. Since the proposed changes will not adversely impact the operation of any safety related component or equipment, there can be no impact on the occurrence of any accident. Furthermore, there is no change in plant testing proposed in this change request that could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to LCO 3.7.7 establish consistent and appropriate requirements for main turbine bypass valve operability requirements. These changes do not impact the assumptions contained in these UFSAR analyses since they do not change the manner in which Hope Creek is currently permitted to operate. Since the ACTION Statement for LCO 3.7.7 already allows indefinite continued operation below 25% of RATED THERMAL POWER with an inoperable main turbine bypass valve system, the proposed modification to the APPLICABILITY statement for this LCO does not involve: (1) hardware changes; (2) significant changes to the operation of any systems or components in normal or accident operating conditions; or (3) changes to existing structures, systems or components. The proposed changes will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: (1) no significant hardware changes; (2) no significant changes to the operation of any systems or components; and (3) no changes to existing structures, systems or components, there can be no impact on the occurrence of any accident. Furthermore, there is no change in plant testing proposed in this change request that could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the HC emergency diesel generator (EDG) TS Bases [Change 5—Bases for TS ¾.8, ''Electrical Power Systems''] include information contained in the Safety Evaluation Report for Technical Specification Amendment No. 75. This information concerns the bases for the allowed-outage-time (AOT) for the C and D EDGs. Concerning the revisions to planned C and D EDG outages, PSE&G believes that implementation of 10CFR50.65 requirements to monitor EDG unavailability will provide an acceptable and more clearly defined method for maintaining EDG availability within acceptable limits. As stated in PSE&G's letter LR–N97167, dated March 21, 1997, Hope Creek will not plan C or D EDG outages that exceed 72 hours if the total unavailability of the EDG will be greater than 720 hours on a 12 month rolling basis. The proposed changes will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: (1) No hardware changes; (2) no significant changes to the operation of any systems or components; and (3) no changes to existing structures, systems or components, there can be no impact on the occurrence of any accident. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will 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 for the TS related to response time testing reflect testing methodologies that were approved by the NRC in Amendment No. 85 to the Hope Creek TS. No changes are being made to methodologies with this proposal. Therefore, the changes contained in this request do not result in a significant reduction in a margin of safety.

The proposed changes to Surveillance Requirement 4.0.5 do not alter the current requirements for the Hope Creek inservice inspection and inservice testing programs and are considered to be editorial in nature. Therefore, the changes contained in this request do not result in a significant reduction in a margin of safety.

The proposed changes to the drywell and suppression chamber purge system are being made to reflect design modifications that have been installed. This design modification replaced isolation valves containing resilient material seals with metal seated valves under 10 CFR 50.59. PSE&G has concluded that the 24 month frequency is appropriate for the new valves since: (1) this frequency is imposed by Surveillance Requirement 4.6.1.2.d, which is applicable to other containment isolation valves in Table 3.6.3–1 that penetrate the primary containment; and (2) concerns raised about severe environment-induced degradation and frequent use for the previously installed resilient seal material valves are not applicable to the replacement metal seat valves. The valve modification was an enhancement to the Hope Creek design that did not impact the isolation capability of the drywell and suppression chamber purge system, and does not result in a significant reduction in a margin of safety.

The proposed changes to LCO 3.7.7 establish consistent and appropriate requirements for main turbine bypass valve operability requirements. These changes do not impact the assumptions contained in these UFSAR analyses since they do not change the manner in which Hope Creek is currently permitted to operate. Since the ACTION Statement for LCO 3.7.7 already allows indefinite continued operation below 25% of RATED THERMAL POWER with an inoperable main turbine bypass valve system, the proposed modification to the APPLICABILITY statement for this LCO would be editorial in nature. Therefore, the changes contained in this request do not result in a significant reduction in a margin of safety.

The HC TS Bases [Change 5—Bases for TS ¾.8, ''Electrical Power Systems''] will be revised to include information contained in the Safety Evaluation Report for Technical Specification Amendment No. 75. This information concerns the bases for the allowed-outage-time (AOT) for the C and D emergency diesel generators (EDGs). PSE&G believes that implementation of 10 CFR 50.65 requirements to monitor EDG unavailability limits will provide an acceptable and more clearly defined method for maintaining EDG availability within acceptable limits and not result in a significant reduction in a margin of safety.

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

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

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

Date of amendment request: May 19, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3.7.1.3, ''Ultimate Heat Sink'' to reflect that continued plant operation depends upon the association of ultimate heat sink (UHS) temperature *and* safety system availability. The requirements of TS 3.7.1.1, ''Safety Auxiliaries Cooling System (SACS)'', TS 3.7.1.2, ''Station Service Water System (SSWS)'' and TS 3.8.1.1, ''Electrical Power Systems'' would be revised to reflect the revised TS 3.7.1.3. In addition, the Bases for ¾.7.1, ''Service Water Systems'' would be appropriately revised.

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

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

The proposed TS revisions related to SSWS/SACS and the emergency diesel generators (EDGs) [TS 3.7.1.1, TS 3.7.1.2, and TS 3.8.1.1] involve no hardware changes and no changes to existing structures, systems or components. The additional system configuration limits and changes to the operation of SSWS/SACS/EDGs are being made to ensure that SSWS/SACS can remove required heat loads during design basis accidents and transients with the proposed UHS river water temperature and level limits. The link to the UHS LCO in the proposed SSWS/SACS/EDG TS ACTION Statements and the proposed revisions to the SACS ACTION Statement for one inoperable SACS subsystem ensure that the plant is directed to enter a safe shutdown condition whenever the capability to mitigate design basis accidents and transients is lost. Since the SSWS/SACS/EDGs will still remain capable of meeting all applicable design basis requirements and retaining the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not increase the probability of an accident previously evaluated nor significantly increase in the consequences of an accident previously evaluated.

The proposed TS revisions related to UHS [TS 3.7.1.3] involve no hardware changes and no changes to existing structures, systems or components. The additional system configuration limits and changes to the operation of UHS supported systems are being made to ensure that the UHS can remove required heat loads during design basis accidents and transients with the proposed UHS river water temperature and level limits. The proposed UHS TS ACTION Statements ensure that the plant is directed to enter a safe shutdown condition whenever the capability to mitigate design basis accidents and transients is lost. The proposed changes to the UHS TS surveillance requirements to increase monitoring of the river water temperature at 82°F adequately ensures that the actions required when river temperatures exceed 85°F are taken as appropriate. Since the UHS will still remain capable of meeting all applicable design basis requirements and retaining the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes were determined to be justified. As a result, these changes will not increase the probability of an accident previously evaluated nor significantly increase in the consequences of an accident previously evaluated.

With the approval of the proposed changes to the SSWS/SACS/EDG/UHS TS, the proposed TS Bases changes are considered to be editorial in nature. As a result, the proposed Bases changes will not increase the probability of an accident previously evaluated nor significantly increase in the consequences of an accident previously evaluated.

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

The proposed changes to the SSWS/SACS/EDG TS contained in this submittal will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve no hardware changes and no changes to existing structures, systems or components, there can be no impact on the potential occurrence of any accident due to new equipment failure modes. The additional system configuration limits and changes to the operation of SSWS/SACS/EDGs imposed by the proposed changes ensure that SSWS/SACS and the UHS can remove required heat loads during design basis accidents and transients with the proposed UHS river water temperature and level limits. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the UHS TS contained in this submittal will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve no hardware changes and no changes to existing structures, systems or components, there can be no impact on the potential occurrence of any accident due to new equipment failure modes. The additional system configuration limits imposed by the proposed UHS LCO ensure that supported systems can remove required heat loads during design basis accidents and transients with the proposed UHS river water temperature and level limits. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. The proposed changes to the UHS TS surveillance requirements to increase monitoring of the river water temperature at 82 °F adequately ensures that the actions required when river temperatures exceed 85 °F are taken as appropriate. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

With the approval of the proposed changes to the SSWS/SACS/EDG UHS TS, the proposed TS Bases changes are considered to be editorial in nature. As a result, the proposed Bases changes will 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 for the TS related to the SSWS/SACS/EDGs establish consistent and appropriate requirements for SSWS/SACS/EDG and UHS operability requirements. The additional system configuration limits and changes to the operation of SSWS/SACS/EDG are being made to ensure that SSWS/SACS can remove required heat loads during design basis accidents and transients with the proposed UHS river water temperature and level limits. The link to the UHS LCO in the proposed SSWS/SACS/EDG TS ACTION Statements and the revision to the SACS ACTION Statement for one inoperable SACS subsystem ensure that the plant is directed to: (1) enter a safe shutdown condition whenever the capability to mitigate design basis accidents and transients is lost; or (2) enter a conservatively short period of continued operation when system redundancy is reduced. Since the SSWS/SACS/EDG will still remain capable of meeting all applicable design basis requirements and retaining the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes contained in this submittal were determined to not result in a significant reduction in a margin of safety.

The proposed changes for the TS related to the UHS ensure continued capability of the UHS to mitigate the consequences of design basis accidents and transients. The additional SSWS/SACS configuration limits and changes to the operating limits of the UHS ensure that the UHS can remove required heat loads during design basis accidents and transients with the proposed river water temperature and level limits. The proposed UHS TS ACTION Statements ensure that the plant is directed to: (1) enter a safe shutdown condition whenever the capability to mitigate design basis accidents and transients is lost; or (2) enter a conservatively short period of continued operation when supported system redundancy is reduced. Since the UHS will still remain capable of meeting all applicable design basis requirements and retaining the capability to mitigate the consequences of accidents described in the HC UFSAR, the proposed changes contained were determined to not result in a significant reduction in a margin of safety.

With the approval of the proposed changes to the SSWS/SACS/UHS TS, the proposed TS Bases changes are considered to be editorial in nature. As a result, the proposed bases changes will not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: 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.

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

Date of amendment request: May 21, 1997

Description of amendment request: The proposed amendment would revise the Virgil C. Summer Nuclear Station Technical Specifications (TS), Surveillance Requirements (SRs), to change the methodology for testing the charcoal adsorbers in (1) the control room normal and emergency air handling system (TS 3/4.7.6), and (2) the spent fuel pool ventilation system (TS 3/4.9.11), by reference to the methodology of ASTM D 3803–1989 from the ANSI STD N509–1980.

The proposed reference testing methodology to ASTM D 3803–1989 for the control room is at a relative humidity of 70% and 30 degrees C with methyl iodide penetration of < 2.5%. The proposed reference testing methodology to ASTM D 3803–1989 for the spent fuel pool is at a relative humidity of 95% and 30 degrees C with a methyl iodide penetration of < 2.5%.

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

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

The proposed change revises the methodology for testing the charcoal adsorbers in the Control Room Normal and Emergency Air Handling System and the Spent fuel Pool Ventilation System (Engineered Safeguards Feature [ESF] air handling units) to the updated Standard Test Method for Nuclear-Grade Carbon.* * *. The charcoal adsorbers are not initiators of any analyzed event.* * * The charcoal adsorbers will be tested to the updated version of the approved standard, which generally contains more stringent testing requirements. The change does not affect the operation of the ESF air handling units. The new testing requirements will continue to ensure that the ESF air handling units will be capable of performing their safety function and meeting the assumptions in the safety analysis [Final Safety Analysis Report (FSAR)]. The change does not affect the mitigation capabilities of any component or system nor does it affect the assumptions relative to the mitigation of accidents or transients. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change revises the methodology for testing the charcoal adsorbers in the Control Room Normal and Emergency Air Handling System and the Spent fuel Pool Ventilation System * * * to the updated Standard Test Method for Nuclear-Grade Carbon. The change does not involve a significant change in the design or operation of the plant. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed), or new or unusual operator actions. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of this change. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change revises the methodology for testing the charcoal adsorbers in the Control Room Normal and Emergency Air Handling System and the Spent fuel Pool Ventilation System * * * to the updated Standard Test Method for Nuclear-Grade Carbon. Testing of the charcoal adsorbers in the ESF air handling units to the new standard will continue to ensure the systems perform their design function. The increase in the allowed penetration percentage does not affect the accident analysis because testing requirements are more stringent and the higher allowed percentages continue to be below the assumptions of the safety analysis [FSAR]. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Gordon Edison, Acting.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: May 27, 1997

Description of amendments request: The proposed amendments would revise the applicable Modes for Source Range Nuclear Instrumentation (Technical Specification 3/4.3.1, ''Reactor Trip System Instrumentation''), provide allowances for an exception to the requirements for the state of the power supplies for Residual Heat Removal System discharge to charging pump suction valves following Mode changes (Technical Specification 3/4.5.2, ''ECCS Subsystems—T_{avg} greater than 350°F'' and 3/4.5.3, ''ECCS Subsystems—T_{avg} less than 350°F''), and delete cycle-specific guidance concerning manual emergency engineered safety feature function input checks.

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 significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The purposes for repositioning the breakers/disconnects for MOVs [motor-operated valves] 8706A and 8706B are to ensure that the ECCS [Emergency Core Cooling System] System is aligned properly such that the assumptions used in the safety analyses are met and to prevent possible overpressurization of the charging pump suction line piping. The likelihood of a severe transient occurring in this time frame is very small and has to be weighed against the possibility of over pressurizing the CVCS [Chemical and Volume Control System] charging pump suction piping. The allowance of a 4 hour time period to perform the required alignment appropriately weighs this risk. Changing the applicability of the requirement to have indication from a Source Range Nuclear Instrument available to agree with the design of the plant does not change the physical design of the plant or affect any assumptions used in accident analyses and, therefore, has no effect on the probability or consequences of an accident previously evaluated in the FSAR. The allowance of 1 hour to perform the Source Range Channel Check upon reaching P–6 from Mode 2 is consistent with the current basis for a source range channel inoperable. Therefore, these changes do not involve a significant increase in the consequences of an accident previously evaluated.

(2) The proposed changes to the Technical Specifications do not increase the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new limiting single failure or accident scenario has been created or identified due to the proposed changes. Safety-related systems will continue to perform as designed. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed changes do not involve a significant reduction in the margin of safety. The margin of safety is not significantly reduced due to the proposed changes to the breaker/disconnect positioning requirements of TS [Technical Specifications] 3/4.5.2 and 3/4.5.3 when transitioning between Modes 3 and 4. The likelihood of either a severe transient occurring in Mode 3 or the possible overpressurization of the CVCS charging pump suction line by the RHR [residual heat removal] system in Mode 4 is very small. Changing the Applicability of the requirement to have indication from a Source Range Nuclear Instrument available to agree with the design of the plant does not change the physical design of the plant or affect any assumptions used in accident analyses and, therefore, has no effect on the margin of safety. These proposed changes are technically consistent with the requirements and standard format of NUREG–1431, Revision 1. Performing the source range channel check within 1 hour upon reaching P–6 from Mode 2 does not change the physical design of the plant or affect any assumptions used in accident analyses and, therefore, also does not [a]ffect the margin of safety. Thus, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Project Director: Herbert N. Berkow.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: May 28, 1997

Description of amendments request: The proposed amendments would insert a footnote in Technical Specification (TS) Surveillance Requirement 4.8.1.1.2.e, to clarify that load rejection testing of the shared emergency diesel generator set on either unit may be used to satisfy TS 4.8.1.1.2.e surveillance requirements for both units.

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 clarify that load rejection testing of the shared emergency diesel generator set is only required once per five years, and that testing of the shared EDG [emergency diesel generator] set on one unit may be used to satisfy SR [Surveillance Requirement] 4.8.1.1.2.e requirements for both units. These changes do not affect the probability or consequences of an accident. There are no changes being made to the emergency diesel generator testing program. These changes simply clarify the existing test program and the intent of the test requirements.

Therefore, the proposed TS 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 clarify that load rejection testing of the shared emergency diesel generator set is only required once per five years, and that testing of the shared EDG set on one unit may be used to satisfy SR 4.8.1.1.2.e requirements for both units. No new testing configuration is being proposed that could create the possibility of any new or different kind of accident from any accident previously evaluated. There are no changes being made to the emergency diesel generator testing program. These changes simply clarify the existing test program and the intent of the test requirements.

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 clarify that load rejection testing of the shared emergency diesel generator set is only required once per five years, and that testing of the shared EDG set on one unit may be used to satisfy SR 4.8.1.1.2.e requirements for both units. A similar technical specification change has been previously approved by the NRC for Hatch Nuclear Plant. The technical specification bases and the Final Safety Analysis Report have been reviewed. Clarification of the testing requirements has no effect on the margin of plant safety since no reduction in the test program is involved.

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

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

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: May 2, 1997.

Description of amendment request: The proposed change would continue to allow entry into Operational Conditions 1, 2, and 3 with the inboard main steam isolation valve (MSIV) leakage control subsystem inoperable.

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

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

This License Amendment application proposes a revision to the exception to Limiting Condition for Operation (LCO) 3.0.4 as it applies to the Technical Specification (TS) for the MSIV Leakage Control System (LCS). This revision is proposed to permit completion of activities necessary to implement the most appropriate permanent resolution for the issues that resulted from the elimination of the secondary containment bypass leakage path through the Main Steam Line drains. In addition, the revision clarifies that the exception only applies to the Inboard MSIV LCS subsystem. The drains will remain in their current configuration, which seals off the secondary containment bypass leakage path. The sealed drain path results in a temporary inoperability of the Inboard MSIV LCS subsystem when the plant is operated below 50 percent rated thermal power (RTP), due to condensate build-up in the bottom of the steam lines between the MSIVs. The requested 3.0.4 exception is necessary to permit plant startups with this temporary inoperability. The exception to LCO 3.0.4 simply permits use of the existing Action statement (Condition A of LCO 3.6.1.9) during MODE changes.

The probability of occurrence of a previously evaluated accident is not affected by the proposed revision of the LCO 3.0.4 exception since no change to the plant or to the manner in which the plant is operated is involved. The existing plant configuration will be maintained, and possible concerns resulting from that configuration have been analyzed. The extra weight of the water pooled between the MSIVs was analyzed with respect to piping supports and seismic considerations and was found to be acceptable, and condensate that is carried past the outboard MSIVs will be drained to the condenser by drain connections downstream of the outboard MSIVs before it can reach the turbine. The temporary inoperability of the Inboard MSIV LCS subsystem when below 50 percent RTP has no impact on accident initiation probability, since the MSIV LCS does not serve to prevent accidents, but is only used in mitigating the consequences of Loss of Coolant Accidents (LOCAs) that have already occurred.

The consequences of an accident are not affected in that the Outboard MSIV LCS subsystem will be available to perform the MSIV LCS function by mitigating the consequences of a LOCA during the temporary period in which the Inboard MSIV LCS subsystem is unavailable. Condensate that is carried past the outboard MSIVs will be drained to the condenser by drain connections downstream of the outboard MSIVs; therefore, no impairment of the Outboard MSIV LCS subsystem will result from condensed water. The Required Action and Completion Time for one inoperable MSIV LCS subsystem remains the same, and limits plant operation to the previously established 30-day Allowable Outage Time. The Required Action if both subsystems of MSIV LCS were to become inoperable also remains the same. The MSIV function of isolating the Main Steam Lines is also unaffected by the existing plant configuration, since MSIV performance will not be affected by the existence of accumulated water in the bottom of the steam lines between the MSIVs during plant operation below 50 percent RTP. Therefore, if necessary, the Main Steam Lines will be isolated, and leakage past the MSIVs will be routed for filtration as in the design-basis radiological analyses, and the safety and radiological consequences of previously evaluated accidents will remain unaffected.

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

The proposed change to permit inoperability of the Inboard MSIV LCS subsystem during periods of startup and power ascension to 50 percent RTP and during shutdown below 50 percent RTP does not create the possibility of a new or different kind of accident from any previously evaluated. The Inboard MSIV LCS subsystem is only credited during a large-break LOCA wherein Reactor Coolant System depressurization occurs. The temporary unavailability of the Inboard MSIV LCS subsystem can be mitigated by operation of the Outboard MSIV LCS subsystem. The amendment to the TS is an administrative change that does not involve change to the current plant design or methods of operation. No new plant equipment failure modes or accident initiators are introduced by the LCO 3.0.4 exception.

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

The response to a large-break LOCA will not be affected since the Outboard MSIV LCS subsystem can be assumed to be available during the limited period of time that the Technical Specifications permit the Inboard subsystem to be unavailable. Allowing entry into MODES 1, 2, and 3 while utilizing the existing Condition A and Required Action A.1 does not reduce the margin of safety since the Completion Time allowed for that Condition is not increased. The proposed change will have no adverse impact on the reactor coolant system pressure boundary nor will other system protective boundaries or safety limits be affected.

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

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

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

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: May 2, 1997

Description of amendment request: The proposed change would allow the leakage rate of one or more main steam lines to be up to 35 standard cubic feet per hour (scfh), as long as the total leakage rate through all four main steam lines is less than or equal to 100 scfh.

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 involves the deletion of the portion of Technical Specification Surveillance Requirement (SR) 3.6.1.3.10 that states the increased leakage rate of less than or equal to 35 scfh for an individual main steam line is only acceptable for Operating Cycle 6, and a deletion of the restriction that a main steam line leakage rate of less than or equal to 35 scfh is acceptable for only one main steam line. The overall main steam line leakage limit of less than or equal to 100 scfh for all four main steam lines is not being revised.

The MSIV [main steam isolation valve] leakage is not an initiator of an accident, including the steam line rupture accident. Therefore, the probability of an accident previously evaluated has not changed.

The consequences of interest are the radiological dose consequences following a large-break Loss of Coolant Accident (LOCA). This is the event which the regulatory guidance requires to be evaluated using the extremely conservative source term assumptions of Regulatory Guide 1.3, ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.'' Since the overall main steam line leakage rate of less than or equal to 100 scfh for all four main steam lines is not being revised, the radiological consequences of an accident previously evaluated has not changed.

Therefore, the probability or consequences of an accident previously evaluated have not significantly increased.

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

This proposed change does not physically alter the plant or systems or equipment in the plant, or the method for operation of the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not revise the overall combined leakage rate of less than or equal to 100 scfh for all four main steam lines that is permitted in the present Specification. It is the combined main steam line penetration leakage rate that is assumed in the radiological accident analyses. Thus, although individual steam line leakage rates may be less than or equal to 35 scfh, as long as overall leakage of the four main steam lines is maintained at its current value of less than or equal to 100 scfh, the proposed change does not reduce the margin of safety.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus.

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: November 9, 1987, as supplemented March 31, 1988, June 8, 1992 and February 4, 1997

Description of amendment request: The proposed changes would revise the Technical Specifications (TS) for the North Anna Power Station (NA 1&2). The changes would reformat the operability and surveillance requirements for the intermediate range (IR) channels to be consistent with NUREG–0452, Revision 4, ''Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors'' (Fall 1981), which is applicable to NA 1&2. Also, the proposed changes would revise the nominal IR high flux trip setpoint. The IR nuclear flux trips provide backup reactor core protection during reactor startup. There is no operating condition under which the IR trip provides sole overpower protection. It is a backup trip only, and no credit is taken for the trip in the NA 1&2 Updated Final Safety Analysis Report (UFSAR). Operating experience at NA 1&2 has shown the IR channel response to be sensitive to core loading patterns, varying core burnups, and control rod positions, and the variability in the channel response had made it difficult to maintain the channels in proper calibration. Therefore, the proposed change would elevate the nominal IR high flux trip setpoint from a current equivalent to 25% of rated thermal power to a current equivalent to 35% of rated thermal power.

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

[The proposed changes would not:]

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. There is no adverse impact on the safety analysis (since no credit is taken for the trips in the existing analyses), and no degradation of the protection system redundancy or reliability. This latter conclusion is based on sensitivity studies which show that the effectiveness of the flux trip system in protecting against the low power reactivity excursions examined in the FSAR is not sensitive to realistic variations in the actual flux trip setpoint.

2. Create the probability of a new or different kind of accident from any accident previously identified, since the severity of the analyzed accidents is unchanged, and since only a change to a setpoint and the associated surveillance requirements for the reactor protection system is involved.

3. Involve a significant reduction in a margin of safety, since none of the safety analysis input or assumptions are changed, nor are the probability nor the consequences of any previously analyzed accidents increased.

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: Brenda Mozafari (Acting).

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.

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

Date of application for amendment: March 31, 1997

Brief description of amendment: The proposed amendment would remove containment isolation valve 863 from Technical Specification Table 3.6–1, ''Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation.''

Date of publication of individual notice in **Federal Register**: May 15, 1997 (62 FR 26823).

Expiration date of individual notice: June 16, 1997.

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

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

Date of amendment request: April 25, 1997

Brief description of amendment request: The proposed amendment changes to revise Technical Specification 3.5.2 to eliminate the flow path from the residual heat removal system to the reactor coolant system hot legs that is specified in Limiting Condition for Operation 3.5.2.c.2.

Date of publication of individual notice in **Federal Register**: May 14, 1997 (62 FR 26574).

Expiration date of individual notice: June 13, 1997.

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

Public Service Electric & Gas Company, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: May 1, 1997

Brief description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.7.7, ''Auxiliary Building Exhaust Air Filtration System,'' and add a new TS Section 3/4.7.11, ''Switchgear and Penetration Area Ventilation System.'' The change to TS 3/4.7.7 would allow for an increase in the allowed outage time from 7 to 14 days when one auxiliary building exhaust fan is inoperable. The new TS 3/4.7.11 addresses the support function this system provides to other necessary safety support components.

Date of publication of individual notice in **Federal Register**: May 15, 1997 (62 FR 26826).

Expiration date of individual notice: June 16, 1997.

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

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

Date of amendment request: May 14, 1997

Brief description of amendment request: Your application proposes changes to revise Technical Specification Surveillance Requirement 4.7.6.1.d.1 to indicate that the specified acceptable filter differential pressure (DP) is to be measured across the filter housing and to change the filter DP acceptance value from less than or equal to 3.5 inches water gauge to less than or equal to 2.70 inches water gauge.

Date of publication of individual notice in **Federal Register**: May 29, 1997 (62 FR 29158).

Expiration date of individual notice: June 30, 1997.

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

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.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: November 26, 1997

Brief description of amendment: The amendment revises Technical Specifications Definition 1.M, ''Primary Containment Integrity,'' Note 6 on Table 3.2.A for the high flow main steam line instrumentation, Table 3.2.D for a typographical error, Table 3.2.F to reflect a change made in instrument type for the suppression chamber water temperature instrumentation, Table 3.2.F to reflect modifications made to suppression chamber bulk and local temperature instrumentation, Bases Section 3/4.6G to remove an obsolete reference to Group I welds, and Bases Section 3/4.7.A to remove ''high radiation'' from the description of Primary Containment Group 1 initiation signals. In addition, this amendment includes changes made to the Bases Section 3.10, ''Core Alterations,'' as noted by BECo letter dated March 7, 1997.

Date of issuance: May 28, 1997.

Effective date: May 28, 1997.

Amendment No.: 172.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6568). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

Date of application for amendments: June 10, 1996, as supplemented by letter dated February 17, 1997

Brief description of amendments: The amendments change the Technical Specifications to reflect the transition from General Electric Company (GE) to Siemens Power Corporation (SPC) as the fuel supplier for the Quad Cities Nuclear Power Station, Units 1 and 2. In addition, as an administrative action by the Commission that only involves the format of the licenses and does not authorize any activities outside the scope of the application and supplement, the NRC has amended the licenses to include an Appendix C that lists additional license conditions. The additional license condition as a result of the review of this application reflects the relocation of the contents of TS 5.4 to the Updated Final Safety Analysis Report.

Date of issuance: May 23, 1997.

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

Amendment Nos.: 177 and 175.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Licenses, Technical Specifications and Updated Final Safety Analaysis Report.

Date of initial notice in **Federal Register**: August 28, 1996 (61 FR 44355). The February 17, 1997, submittal provided additional 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 May 23, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 29, 1995, as supplemented August 7, 1996, and January 10, 1997

Brief description of amendment: The amendment revises Technical Specifications to incorporate the commitments made in connection with Amendment No. 183, which allowed the installation of laser welded sleeves inside of defective steam generator tubes.

Date of issuance: May 20, 1997.

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

Amendment No.: 192.

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

Date of initial notice in **Federal Register**: November 8, 1995 (60 FR 56365) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 1997.

No significant hazards consideration comments received: No.

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

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

Dates of application for amendment: December 24, 1996, and January 31, 1997

Brief description of amendment: Changes Administrative Controls Section of the Technical Specifications to implement revised management responsibilities and titles that reflect the permanently shut down status of the plant.

Date of issuance: May 22, 1997.

Effective date: Effective May 22, 1997, to be implemented within 60 days of issuance.

Amendment No.: 191.

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

Date of initial notice in **Federal Register**: March 26, 1997 (62 FR 14460) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 10, 1997

Brief description of amendments: These amendments modify Unit No. 1 Technical Specification (TS) 5.2.1 to add ZIRLO as fuel assembly material and add reference to the Nuclear Regulatory Commission approved Topical Report WCAP–12610, ''Vantage+ Fuel Assembly Reference Core Report,'' to TS 6.9.1.12 for both units.

Date of issuance: May 23, 1997.

Effective date: Both units, as of date of issuance, to be implemented within 60 days.

Amendment Nos.: 203 and 84.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17231) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 23, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: February 5, 1997, as supplemented by letter dated March 26, 1997

Brief description of amendment: The amendment changes the Appendix A Technical Specifications for Waterford Steam Electric Station, Unit 3, by revising Technical Specifications 3.1.2.7, 3.1.2.8, 3.5.1, 3.5.4, 3.9.1, and Bases 3/4.1.2. The changes will increase the minimum boron concentration in the Safety Injection Tanks and the Refueling Water Storage Pool from 1720 to 2050 ppm.

Date of issuance: May 29, 1997, to be implemented within 60 days.

Effective date: May 29, 1997.

Amendment No.: 129.

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

Date of initial notice in **Federal Register**: March 26, 1997, (62 FR 14461) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 29, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 28, 1996, as supplemented March 11, 1997

Brief description of amendment: The amendment revises Three Mile Island, Unit 1, Technical Specifications to permit the use of 10 CFR 50, Appendix J, Option B, Performance-Based Containment Leakage Testing.

Date of issuance: May 27, 1997.

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

Amendment No.: 201.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 31, 1996 (61 FR 40019) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 27, 1997.

No significant hazards consideration comments received: No.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: August 8, 1996

Brief description of amendments: The amendments allowed the transition from Mode 4 to Mode 3 with the turbine-driven auxiliary feedwater pump inoperable and allowed a 72-hour period after the entry into Mode 3 to complete all necessary operability testing.

Date of issuance: May 27, 1997.

Effective date: May 27, 1997, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 87; Unit 2—Amendment No. 74.

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

Date of initial notice in **Federal Register**: August 28, 1996 (61 FR 44359) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 27, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 6, 1997

Brief description of amendment: The amendment revises the Technical Specifications on allowed outage times for certain protective instrumentation and also for reactor building access control. The amendment adopts, in part, guidance from NUREG–0123, ''Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5),'' Revision 3, and NUREG–1433, ''Standard Technical Specifications General Electric Plants BWR/4,'' Revision 1.

Date of issuance: May 28, 1997.

Effective date: As of the date of issuance, to be implemented within 90 days.

Amendment No.: 101.

Facility Operating License No. DPR–21: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 26, 1997 (62 FR 14462) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360 and at the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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

Date of application for amendment: March 31, 1997

Brief description of amendment: The amendment modifies Technical Specification Surveillance 4.7.1.2.1.b, which requires the testing of the auxiliary feedwater motor-driven and turbine-driven pumps on recirculation flow at least once per 92 days. The amendment also makes changes to the appropriate Bases section.

Date of issuance: May 29, 1997.

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

Amendment No.: 139.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19832) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 29, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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

Date of application for amendment: March 31, 1997

Brief description of amendment: The amendment separates the required testing of motor-operated valve thermal overload protection into two new surveillances.

Date of issuance: May 29, 1997.

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

Amendment No.: 140.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19833) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 29, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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

Date of application for amendment: January 16, 1997, as supplemented on February 24, 1997

Brief description of amendment: This amendment revises the license to delete the prohibition on moving a spent fuel assembly shipping cask into the Fuel Building.

Date of issuance: May 19, 1997.

Effective date: This license amendment is effective as of the date of issuance (May 19, 1997), but shall be implemented within 30 days of issuance.

Amendment No.: 196.

Facility Operating License No. NPF–1: The amendment revised the license.

Date of initial notice in **Federal Register**: March 26, 1997 (62 FR 14467).

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 28, 1997

Brief description of amendment: This amendment changes the Permanently Defueled Technical Specifications to delete the requirement for NRC prior approval to changes in the Certified Fuel Handler's Training Program.

Date of issuance: May 23, 1997.

Effective date: May 23, 1997.

Amendment No.: 197.

Possession-Only License No. NPF–1: The amendment revised the Technical Specifications.

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

No significant hazards consideration comments received: No.

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

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: April 15, 1997

Brief description of amendments: These amendments revise Surviellance Requirement 3.8.1.8 of Technical Specifications (TS) 3.8.1, ''AC Sources—Operating,'' for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The TS change will allow the licensee to credit overlap testing to validate the capability of the alternate offsite power source.

Date of issuance: June 2, 1997.

Effective date: June 2, 1997.

Amendment Nos.: Unit 2—136; Unit 3—128.

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

Date of initial notice in **Federal Register**: May 1, 1997 (62 FR 23811) The Commission's related evaluation of the amendments is contained in a Safety E valuation dated June 2, 1997.

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 No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: January 10, 1997, as supplemented May 2 and May 15, 1997

Brief description of amendment: The amendment modifies the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications (TS) in order to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

Date of issuance: May 27, 1997.

Effective date: May 27, 1997.

Amendment No.: 5.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: January 29, 1997 (62 FR 4356) The May 2 and May 15, 1997 letters 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 May 27, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

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

Date of amendment request: March 18, 1997

Brief description of amendment: This amendment revises Technical Specification Surveillance Requirement 4.5.2.c to clarify when a containment entry visual inspection is required. This change reduces the visual inspection requirement to at least once daily and is in accordance with the guidance provided in Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Date of issuance: May 28, 1997.

Effective date: May 28, 1997, to be implemented within 30 days of the date of issuance.

Amendment No.: 105.

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

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19839) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 1997.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By July 18, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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

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

Commonwealth Edison Company, Docket No. STN 50–456, Braidwood Station, Unit No. 1, Will County, Illinois

Date of application for amendment: Two submittals dated May 23, 1997

Brief description of amendment: The amendment revises Technical Specification (TS) 4.5.2.b.1 to include the use of ultrasonic testing (UT) to verify that the emergency core cooling system (ECCS) is completely filled with water. For the ECCS subsystems with high point vent valves in direct communication with the operating systems, UT is acceptable in lieu of physically opening the vents.

Date of Issuance: May 23, 1997.

Effective date: Immediately, to be implemented within 30 days.

Amendment No.: 83.

Facility Operating License No. NPF–72: The amendment revised the TSs.

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 23, 1997.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket No. STN 50–454, Byron Station, Unit No. 1, Ogle County, Illinois

Date of application for amendment: May 24, 1997, as supplemented on May 31, 1997

Brief description of amendment: The amendment revises Technical Specification 4.5.2.b.1 to include the use of ultrasonic testing (UT) to verify that the emergency core cooling system (ECCS) is completely filled with water. For the ECCS subsystems with high point vent valves in direct communication with the operating systems, UT is acceptable in lieu of physically opening the vents. This amendment supersedes NOED No. 97–6–010 for Byron, Unit 1, which was granted on May 23, 1997.

Date of Issuance: June 1, 1997.

Effective date: Immediately, to be implemented within 30 days.

Amendment No.: 90.

Facility Operating License No. NPF–37: The amendment revised the TS. Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated June 1, 1997.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

Local Public Document Room location: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

NRC Project Director: Robert A. Capra.

Dated at Rockville, Maryland, this 11th day of June, 1997.

For The Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 97–15827 Filed 6–17–97; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

Changes in Domestic Mail Rates and Classifications

AGENCY: Postal Service.

ACTION: Notice of implementation of changes in domestic mail rates for Classroom Periodicals.

SUMMARY: This notice sets forth the changes in permanent rates for Classroom Periodicals to be implemented as a result of a decision of the Governors of the Postal Service in Docket No. MC96–2, and the resulting changes in temporary rates for Classroom Periodicals to be implemented concurrent with the movement to the next step of phasing.

EFFECTIVE DATE: October 5, 1997.

FOR FURTHER INFORMATION CONTACT: Eric Koetting, (202) 268–2992.

SUPPLEMENTARY INFORMATION: On April 4, 1996, pursuant to its authority under 39 U.S.C. 3621 *et seq.*, the Postal Service filed with the Postal Rate Commission (PRC) a request for a recommended decision on a number of mail classification reform proposals regarding certain types of preferred rate mail (''Classification Reform II (Nonprofit Mail)'', PRC Docket No. MC96–2). The PRC published a notice in the **Federal Register** on April 11, 1996 (61 FR 16129–16146) describing the Postal Service's request and offering interested parties an opportunity to intervene.

On July 19, 1996, the PRC issued its first Opinion and Recommended Decision in Docket No. MC96–2. The PRC's recommendations very closely tracked the Postal Service's proposals, with the exception that the Commission deferred action on the changes proposed regarding the Classroom subclass of Periodicals mail. On August 5, 1996, the Governors of the Postal Service, pursuant to their authority under 39 U.S.C. 3625, approved the permanent